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Volume 4 of 5

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6.0 TRANSPORTATION OF RADIOACTIVE FUEL CYCLE WASTES

6.0 TRANSPORTATION OF RADIOACTIVE FUEL CYCLE WASTES

This section gives a general analysis of transportation requirements for postfission radioactive wastes that are produced from the commercial light water reactor (LWR) fuel cycle and that are assumed to require Federal custody for storage or disposal. Possible radioactive wastes for which transportation requirements are described include:

- spent fuel
- solidified high-level waste
- fuel residues (cladding wastes)
- plutonium
- non-high-level transuranic (TRU) wastes.

Transportation is described for wastes generated in three fuel cycle options:

- *once-through fuel cycle*
- uranium recycle only
- recycle of uranium and plutonium.

Figures 6.1 to 6.3 show schematically the waste transportation requirements for each option.

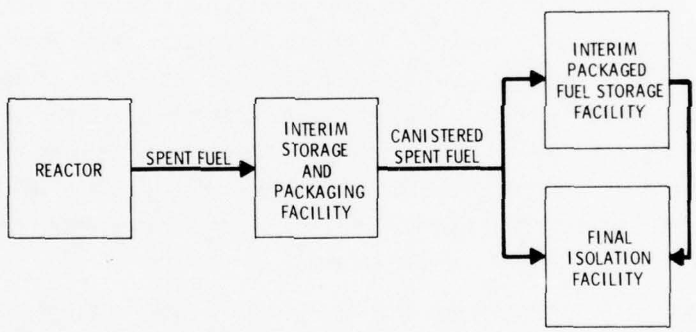

FIGURE 6.1. Transportation of Wastes that are Assumed to Require Custody for Once-Through Fuel Cycle Option

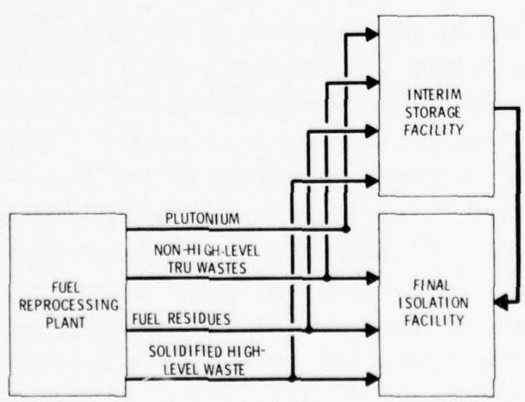

FIGURE 6.2. Transportation of Wastes that are Assumed to Require Custody for Uranium Recycle Only Option

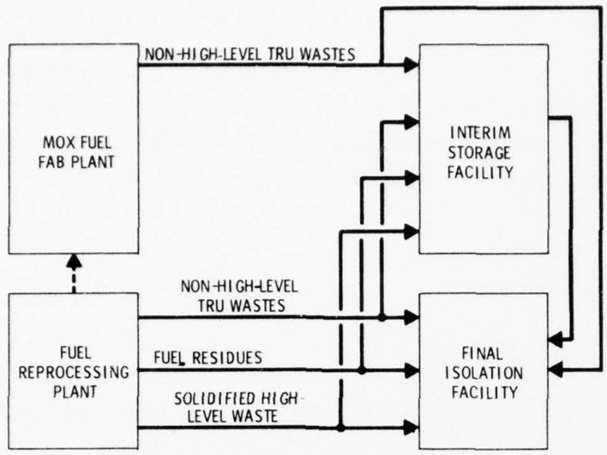

FIGURE 6.3. Transportation of Wastes that are Assumed to Require Federal Custody for Fuel Cycle Option Involving Recycle of Uranium and Plutonium

Shipments of spent fuel have been made commercially on a routine basis for many years. Current technology appears to be adequate for these shipments. Non-high-level TRU wastes are presently shipped by rail from Department of Energy (DOE) facilities to interim storage at the Idaho National Engineering Laboratory (INEL). Waste containers and overpacks for rail and truck shipment of TRU wastes are commercially available. Shipping containers are also presently available for those forms of plutonium that require little or no shielding and have a relatively low heat generation rate. Plutonium shipments are made in accordance with Federal regulations for the physical protection of special nuclear material in transit.

There is presently no commercial processing of spent fuel in the United States, and commercial shipments of solidified high-level waste and of fuel residues have not been made. However, no technological reasons exist which would prevent the design and construction of shipping containers for these wastes. It is expected that shipping casks for solidified high-level waste and for fuel residues will resemble those for spent fuel.

Radioactive waste shipments are expected to be made by commercial carrier. Except for plutonium, which must be shipped in accordance with special procedures outlined in the Code of Federal Regulations (10 CFR 73),[1] these shipments will use conventional transportation equipment. The principal transport modes are truck and rail. Because of legal weight restrictions on the nation's highways, truck shipments are limited to payload weights of about 22 MT (48,000 lb) without overweight permits. Rail shipments can accommodate payloads of about 100 MT (220,000 lb). All radioactive waste shipments must comply with Federal transportation regulations that establish container specifications, radiation dose rate limits, and handling procedures intended to insure the safety of the public and of transportation workers. Containers for radioactive waste must also conform to design limits imposed by the facility receiving the waste.

All waste shipments described in this chapter are expected to be made in shipping containers that meet Type B (see section 6.1.1.4) package performance requirements. Type B performance tests simulate accident environments likely to be encountered in severe transportation accidents. While it is probably impossible to design a container that would withstand all conceivable accident situations, Type B containers will withstand credible accidents.

Table 6.1 summarizes packaging and shipping information on postfission radioactive wastes from the commercial LWR fuel cycle that are assumed to be transported to Federal storage or isolation sites.

TABLE 6.1. Waste Packaging and Shipping Information

Waste	Origin	Destination	Form of Waste	Reference Transport Mode	Packaging
Spent fuel	Reactor	Interim storage and packaging facility	Bare fuel assemblies	Rail and truck	Rail cask or truck cask
Spent fuel	Interim storage and packaging facility	Interim packaged fuel storage facility or final isolation facility	Fuel assemblies in canisters	Rail	Rail cask modified to accept canistered fuel elements
Solidified high-level waste	Fuel reprocessing plant	Interim storage facility or final isolation facility	Borosilicate glass or calcine	Rail	3.05 m long stainless steel canisters; Canister diameters range from 0.15 m to 0.30 m and number of canisters per cask range from 9 to 36
Fuel residues	Fuel reprocessing plant	Interim storage facility or final isolation facility	Uncompacted mixed with sand or treated by mechanical compaction or melted and cast into ingots	Rail	0.76 m x 3.05 m canister, 3 canisters per cask
Plutonium	Fuel reprocessing plant	Interim storage facility or final isolation facility	PuO_2 powder	Truck	Canisters in primary pressure vessel inside Type B cask. Ten casks in special truck trailer.
Non-high-level TRU waste	Fuel reprocessing plant or MOX plant	Interim storage facility or final isolation facility	General trash and noncombustible waste	Truck or rail	55-gal drums. Drums may require a liner. Shipment in reusable Type B overpack.[a]
Non-high-level TRU waste	Fuel reprocessing plant or MOX plant	Interim storage facility or final isolation facility	Wet wastes, particulate solids, incinerator ash and filter media immobilized in cement or bitumen	Truck or rail	55-gal drums. Shipment in reusable Type B overpack.[a]
Non-high-level TRU waste	Fuel reprocessing plant or MOX plant	Interim storage facility or final isolation facility	Failed equipment and noncombustible waste	Truck or rail	55-gal drums and steel boxes shipped in reusable Type B overpacks.[a] Intermediate-level waste shipped in fuel residues canister and cask.

a. Because individual waste packages of TRU waste are assumed to exceed the 0.001 Ci limitation for Group I radionuclides, these shipments are presumed to be made in overpacks that meet Type B package standards or their equivalent. See Sections 6.1.1.1, 6.1.1.4, and 6.6.

6.1 BACKGROUND INFORMATION ON WASTE TRANSPORTATION

6.1 BACKGROUND INFORMATION ON WASTE TRANSPORTATION

6.1.1 Regulations Governing Radioactive Materials Transport

All shipments containing nonexempt* quantities of radioactive material are regulated by the U.S. Department of Transportation (DOT) and the Nuclear Regulatory Commission (NRC). Standards and criteria for packaging and transportation are contained in Title 49 and Title 10 of the Code of Federal Regulations.[2] Federal regulations prescribe shipping container requirements, limitations on package contents, and packaging and handling procedures. Their purpose is to ensure that radioactive material shipments pose minimum risks to the public and to transportation workers. A brief summary of some significant Federal regulations follows.

6.1.1.1 Classification of Radioactive Materials for Shipment

A potential risk to public health and safety exists in a release of radioactive material from a shipping container, from both the external radiation emitted by the exposed radionuclides and from the often more serious problem of ingestion of radioactivity into the body, particularly through inhalation. Since radiotoxicities of different radionuclides vary by several orders of magnitude, present DOT standards take radionuclide toxicity into account.

Each type of radioactive material is classified for transportation purposes into one of seven transport groups according to its potential hazard if released to the environment. Transport Group I is the most restrictive. Plutonium and other transuranic elements are in this transport group. Transport Group VII is the least restrictive. Materials such as tritium gas, with a low potential for producing radiation exposure, are placed in this group. The transport grouping of radionuclides is shown in Table 6.1.1, derived from Appendix C of 10 CFR 71.

For radioactive material belonging to a specific transport group, the type of packaging required depends on both the specific activity and the total quantity of material (i.e., the number of curies) being shipped. Classification of radioactive shipments in terms of the number of curies being shipped is done in four categories: exempt, Type A, Type B, or large quantity. Table 6.1.2 lists the curie limits associated with each category for radioisotopes in each of the seven transport groups.

6.1.1.2 Low Specific-Activity Material

Shipments which pose a negligible risk to public health may be classified as low-specific-activity (LSA) material [10 CFR 71.4 (g)]. If the radioactivity is essentially uniform in distribution, with a concentration of not more than 0.1 μCi/g of Group I material or 5 μCi/g of Group II material or 300 μCi/g of Group III or IV material, the waste qualifies as low-specific-activity material. Externally contaminated nonradioactive materials may be considered as low-specific-activity provided that the radioactive contamination averaged over 1 m^2 does not exceed 0.1 $\mu Ci/cm^2$ for Group I radionuclides or 1.0 $\mu Ci/cm^2$ for others.

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* Some radioactive shipments are exempt from the packaging and labeling requirements of Title 49, Part 173 of the Code of Federal Regulations. Exemption is based on quantity (see Table 6.1.1) and on several other conditions outlined in 49 CFR 173.391.

TABLE 6.1.1. Transport Grouping of Radionuclides

Element[a]	Radionuclide[b]
Transport Group I	
Actinium (89)	Ac 227
	Ac 228
Americium (95)	Am 241
	Am 243
Berkelium (97)	Bk 249
Californium (98)	Cf 249
	Cf 250
	Cf 252
Curium (96)	Cm 242
	Cm 243
	Cm 244
	Cm 245
	Cm 246
Neptunium (93)	Np 237
	Np 239
Plutonium (94)	Pu 238
	Pu 239
	Pu 240
	Pu 241
	Pu 242
Polonium (84)	Po 210
Protactinium (91)	Pa 230
	Pa 231
Radium (88)	Ra 226
	Ra 228
Thorium (90)	Th 228
	Th 230
	Th 231
Uranium (92)	U 232
Transport Group II	
Argon (18)	Ar 41
Barium (56)	Ba 133
Bismuth (83)	Bi 210
Europium (63)	Eu 154
Krypton (36)	Kr 87
Lead (82)	Pb 210
	Pb 212
Mixed Fission Products	
Protactinium (91)	Pa 233
Radium (88)	Ra 223
	Ra 224
Radon (86)	Rn 222
Strontium (38)	Sr 90
Thorium (90)	Th 227
	Th 234
Uranium (92)	U 230
	U 233
	U 234
	U 236
Xenon (54)	Xe 135
Transport Group III	
Antimony (51)	Sb 124
	Sb 125
Astatine (85)	At 211
Barium (56)	Ba 140
Bismuth (83)	Bi 207
	Bi 212
Cadmium (48)	Cd 115m
Cerium (58)	Ce 144
Cesium (55)	Cs 134m
	Cs 134
	Cs 137
Chlorine (17)	Cl 36
Cobalt (27)	Co 56
	Co 60
Dysprosium (66)	Dy 154
Europium (63)	Eu 150
	Eu 152
Gallium (31)	Ga 67
Gold (79)	Au 193
	Au 194
	Au 195
Transport Group III (contd)	
Indium (49)	In 114m
Iodine (53)	I 124
	I 125
	I 126
	I 129
	I 131
	I 133
Iridium (77)	Ir 192
Krypton (36)	Kr 85m
	Kr 85
Lutecium (71)	Lu 172
Magnesium (12)	Mg 28
Nickel (28)	Ni 56
Potassium (19)	K 43
Ruthenium (44)	Ru 106
Samarium (62)	Sm 145
	Sm 147
Scandium (21)	Sc 46
Silver (47)	Ag 110m
Sodium (11)	Na 22
Strontium (38)	Sr 89
	Sr 91
Tantalum (73)	Ta 182
Tellurium (52)	Te 129m
	Te 131m
Terbium (65)	Tb 160
Thallium (81)	Tl 204
Thorium (90)	Th 232
	Th Natural
Thulium (69)	Tm 168
	Tm 170
Tin (50)	Sn 117m
	Sn 121
Uranium (92)	U 235
	U 238
	U Natural
	U Enriched
	U Depleted
Vanadium (23)	V 49
Xenon (54)	Xe 125
	Xe 131m
	Xe 133
Yttrium (39)	Y 88
	Y 91m
	Y 91
Zirconium (40)	Zr 95
Transport Group IV	
Antimony (51)	Sb 122
Arsenic (33)	As 73
	As 74
	As 76
	As 77
Barium (56)	Ba 131
Beryllium (4)	Be 7
Bismuth (83)	Bi 206
Bromine (35)	Br 82
Cadmium (48)	Cd 109
	Cd 115
Calcium (20)	Ca 45
	Ca 47
Carbon (6)	C 14
Cerium (58)	Ce 141
	Ce 143
Cesium (55)	Cs 131
	Cs 135
	Cs 136
Chlorine (17)	Cl 38
Chromium (24)	Cr 51
Cobalt (27)	Co 57
	Co 58m
	Co 58
Copper (29)	Cu 64
Dysprosium (66)	Dy 165
	Dy 166
Transport Group IV (contd)	
Erbium (68)	Er 169
	Er 171
Europium (63)	Eu 152m
	Eu 155
Fluorine (9)	F 18
Gadolinium (64)	Gd 153
	Gd 159
Gallium (31)	Ga 72
Germanium (32)	Ge 71
Gold (79)	Au 196
	Au 198
	Au 199
Hafnium (72)	Hf 181
Holmium (67)	Ho 166
Indium (49)	In 113m
	In 115m
	In 115
Iodine (53)	I 132
	I 134
	I 135
Iridium (77)	Ir 190
	Ir 194
Iron (26)	Fe 55
	Fe 59
Lanthanum (57)	La 140
Lead (82)	Pb 203
Lutecium (71)	Lu 177
Manganese (25)	Mn 52
	Mn 54
	Mn 56
Mercury (80)	Hg 197m
	Hg 197
	Hg 203
Molybdenum (42)	Mo 99
Neodymium (60)	Nd 147
	Nd 149
Nickel (28)	Ni 59
	Ni 63
	Ni 65
Niobium (41)	Nb 93m
	Nb 95
	Nb 97
Osmium (76)	Os 185
	Os 191m
	Os 191
	Os 193
Palladium (46)	Pd 103
	Pd 109
Phosphorus (15)	P 32
Platinum (78)	Pt 191
	Pt 193
	Pt 193m
	Pt 197m
	Pt 197
Potassium (19)	K 42
Praseodymium (59)	Pr 142
	Pr 143
Promethium (61)	Pm 147
	Pm 149
Radon (86)	Rn 220
Rhenium (75)	Re 183
	Re 186
	R3 187
	Re 188
	Re Natural
Rhodium (45)	Rh 103m
	Rh 105
Rubidium (37)	Rb 86
	Rb 87
	Rb Natural
Ruthenium (44)	Ru 97
	Ru 103
	Ru 105
Samarium (62)	Sm 151
	Sm 153
Transport Group IV (contd)	
Scandium (21)	Sc 47
	Sc 48
Selenium (34)	Se 75
Silicon (14)	Si 31
Silver (47)	Ag 105
	Ag 111
Sodium (11)	Na 24
Strontium (38)	Sr 85m
	Sr 85
	Sr 92
Sulfur (16)	S 35
Technetium (43)	Tc 96m
	Tc 96
	Tc 97m
	Tc 97
	Tc 99m
	Tc 99
Tellurium (52)	Te 125m
	Te 127m
	Te 127
	Te 129
	Te 132
Thallium (81)	Tl 200
	Tl 201
	Tl 202
Thulium (69)	Tm 171
Tin (50)	Sn 113
	Sn 125
Tritium (1)	H 3
Tungsten (74)	W 181
	W 185
	W 187
Vanadium (23)	V 48
Ytterbium (70)	Yb 175
Yttrium (39)	Y 90
	Y 92
	Y 93
Zinc (30)	Zn 65
	Zn 69m
	Zn 69
Zirconium (40)	Zr 93
	Zr 97
Transport Group V	
Argon (18)	Ar 41 (uncompressed)
Krypton (36)	Kr 85m (uncompressed)
	Kr 87 (uncompressed)
Xenon (54)	Xe 131m (uncompressed)
	Xe 135 (uncompressed)
Transport Group VI	
Argon (18)	Ar 37
Krypton (36)	Kr 85 (uncompressed)
Xenon (54)	Xe 133 (uncompressed)
Transport Group VII	
Tritium (1)	H 3 (as a gas or absorbed on solid material)

a. Atomic number shown in parentheses.
b. Atomic weight shown after the radionuclide symbol.

TABLE 6.1.2. Quantity Limits for the Seven Transport Groups

Transport Group	Activity Level, Ci			
	Exempt	Type A	Type B	Large Quantity
I	$\leq 10^{-5}$	$>10^{-5}$ to 10^{-3}	$>10^{-3}$ to 20	>20
II	$\leq 10^{-4}$	$>10^{-4}$ to 5×10^{-2}	>0.05 to 20	>20
III	$\leq 10^{-3}$	$>10^{-3}$ to 3	>3 to 200	>200
IV	$\leq 10^{-3}$	$>10^{-3}$ to 20	>20 to 200	>200
V	$\leq 10^{-3}$	$>10^{-3}$ to 20	>20 to 5000	>5000
VI	$\leq 10^{-3}$	$>10^{-3}$ to 10^{3}	$>10^{3}$ to 5×10^{4}	$>5 \times 10^{4}$
VII	≤25	>25 to 10^{3}	$>10^{3}$ to 5×10^{4}	$>5 \times 10^{4}$

Low-specific-activity material shipped in exclusive use vehicles is exempt from most of the packaging requirements of Title 49 of the Code of Federal Regulations [see 49 CFR 173.392 (b)]. Basically only strong, tight packaging that will not leak in normal transport is required. The radiation dose rate limits of 49 CFR 173.393 are still applicable, however.

6.1.1.3 Type A Packaging Standards

If the specific activity of a shipment exceeds LSA limits but the total number of curies is within Type A limits, then Type A packaging is required for the transport of the material. In accordance with DOT specification 7A (49 CFR 178.350), a Type A container must meet the standard construction requirements for packages in 49 CFR 173.24 and 49 CFR 173.393. The package must also prevent loss or dispersal of its radioactive contents and retain its radioactive shielding efficiency if subjected to a series of tests designed to simulate package stresses associated with normal transport. The Type A packaging test series is outlined in 49 CFR 173.398(b) and includes impact, puncture and compression tests, a vibration test, a water spray test, pressure tests and thermal tests. Typical Type A packaging includes steel drums (for example, the specification 17C 55-gal drum), wooden boxes, and steel boxes.(3)

6.1.1.4 Type B Packaging Standards

Type B quantities of radioactive material must be shipped in Type B packaging. Type B packaging includes specification 6M containers,* packaging authorized by NRC, and packaging meeting the 1973 International Atomic Energy Agency (IAEA) requirements (provided DOT has validated the approval of the foreign competent authority). In addition to meeting the standards for a Type A package, a Type B package must be able to survive a series of hypothetical accident test conditions with essentially no loss of containment and limited loss of shielding capability (see 10 CFR 71.36). The test sequence for Type B packages is designed to simulate the damage that might be expected in a severe accident situation. The damage test sequence includes:

- a free fall from a height of 9.2 m (30 ft) onto a flat, essentially unyielding surface, with the package striking in the orientation that does the most damage

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* The specification 6M container is described in 49 CFR 178.104. It consists of an inner container fixed within an outer shell by a prescribed solid centering medium.

- a free fall from a height of 1 m (40 in.) onto a 15.2 cm (6-in.)-dia steel plunger that is long enough, and with the package in the correct orientation, to do maximum damage
- heat input from exposure for 30 min to a fire or other radiant environment having a temperature of 802°C (1475°F) and an emissivity of 0.9
- for fissile material packages only, immersion in water at a depth of 91 cm (3 ft) for 8 hr.

Packaging for Large Quantities. Large quantities of radioactive material are required to be transported in Type B packaging (10 CFR 71.32).

6.1.1.5 Nuclear Criticality Safety

Fissile material (^{233}U, ^{235}U, and Pu) requires control in transport to assure safety from accidental criticality. Not only must the contents of a package be subcritical when delivered to a carrier, they must remain subcritical under all conditions likely to be encountered in transport, including accidents. Nuclear criticality during shipment may be prevented by a limitation of package contents, by package design, or by special precautions or special administrative or operational controls imposed upon transport of the consignment. Fissile material must be packaged and shipped in such a way as to remain subcritical even if the package or shipment were completely immersed in water.* Operational control to prevent criticality during transport may be achieved by assigning a transport index to packages of fissile material and by restricting the total transport index (and hence the total number of packages) in the shipment. NRC regulations in 10 CFR 71.33-.40 specify criteria for evaluating package design and administrative procedures aimed at preventing nuclear criticality in transport.

6.1.1.6 Radiation Dose Rate Limitations

Waste shipments in the nuclear fuel cycle will normally be made in exclusive use vehicles. Department of Transportation regulations (49 CFR 173.393) set the following limits on radiation dose rates associated with exclusive use vehicle shipments:

- 1000 mrem/hr at 0.91 m (3 ft) from the external surface of the package (provided the package is transported in a closed vehicle)
- 200 mrem/hr at the external surface of the vehicle
- 10 mrem/hr at any point 2 m (6 ft) from the vehicle
- 2 mrem/hr at any normally occupied position in the vehicle.

6.1.1.7 Surface-Contamination Levels

DOT regulations [49 CFR 173.393 (h)] require the absence of significant removable surface radioactive contamination on the external accessible surfaces of packages when they are shipped. Levels of removable contamination on the surfaces are determined by a wipe test. The regulations consider that the level is "not significant" if the radioactivity on the wipe does not

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* Packaging to prevent criticality may include special container shapes (for example, a cylinder with restricted diameter-to-length ratio) or a special birdcage design which prevents two packages of fissile material closely contacting each other. The 6M and LLD-1 containers[4] are examples of packaging that includes design features to prevent criticality.

exceed 10^{-11} Ci/cm^2 for beta-gamma emitters and 10^{-12} Ci/cm^2 for alpha emitters.

6.1.1.8 External Temperature

DOT regulations [49 CFR 173.393 (e)] limit the temperature at any accessible surface of a shipping cask to not more than 50°C (122°F) at any time during transport; full-load or "exclusive-use" shipments, however, may have temperatures as high as 82°C (180°F).

6.1.1.9 Shipment of Radioactive Materials with Other Hazardous Materials

DOT regulations (49 CFR 174.538) prohibit the loading, transportation, and storage of radioactive materials together with Class A explosives. (Class A explosives include black powder, propellant explosives, initiating or priming explosives, blasting caps, certain types of explosive projectiles, and detonating fuses.)

6.1.2 Accident Experience with Radioactive Materials Shipments

The current number of shipments of radioactive material in the United States is estimated(5,6) at between 1 million and 2.5 million shipments per year. Most involve small quantities of nuclear isotopes for use in industry, medicine, agriculture, and education. An estimated(7) 50,000 shipments per year are made in Type B packaging. The remainder use LSA or Type A packaging.

Only a small fraction of shipments of radioactive material (less than 1%) currently involve spent fuel and nuclear waste from LWR operation. About 5000 shipments of non-TRU non-high-level waste are made annually by commercial carrier from reactor and fuel fabrication sites to shallow burial grounds.

Accident experience with shipments of radioactive material has been reported by McCluggage(8) for the years 1949-1970 and by Grella(6) for the years 1971-1975.

For the period 1949-1970, McCluggage reported a total of 160 incidents related to transport of nuclear materials. About half of these incidents involved either defective packaging or handling accidents. In 26 of the reported incidents some radioactive material was released beyond the confines of the vehicle, but in only three cases was there aerial dispersal of material and in only one case was radioactive material released to a watercourse.

Since 1971, transportation incidents involving radioactive material have been reported to the Department of Transportation as part of a hazardous materials incident reporting system. Carriers of hazardous materials are required to report incidents based on certain criteria; these include death, personal injury, property damage, and in the case of radioactive material, suspected radioactive contamination. Of the 32,000 hazardous materials incidents reported to DOT during the period 1971-1975, 144 involved transport of radioactive materials. Of these, 74 reports came from air carriers and air freight forwarders, 65 from highway carriers, and 5 from rail carriers.

Most air shipment incidents involved small Type A or exempt packages of radiopharmaceuticals. The most frequent single cause of package failure (39% of incidents reported) was the rolling or falling of small packages from cargo handling carts. Failure to follow proper packaging requirements and procedures was also noted as an important contributor to air shipment incidents.

Highway incident reports covered numerous packaging and material types, ranging from small Type A pharmaceuticals to large casks, drummed LSA wastes, radiography devices, etc. Vehicular accidents accounted for 19% (12 incidents) of the highway incident reports but for only 1 of the 22 reported material releases. By contrast, package integrity criteria such as loose or defective fittings and closures, corrosion, seam failures, etc., were cited in 30% (19 incidents) of the highway reports and in 15 of the 22 reported releases. Other highway incident causes included handling and loading accidents, damage caused by other freight, and 10 instances of vehicular contamination reported by one carrier who reported contamination in dedicated vehicles used for full-load shipments.

Five incident reports involving radioactive materials were submitted by rail carriers. Only two of the five reported incidents resulted in a release of material, and both involved low-specific-activity material.

During the past 25 years, an estimated[9] 3600 packages of irradiated fuel have been transported in routine commerce. All shipments of spent fuel are made in Type B packages. Type B containers will also be used for commercial shipment of high-level waste and fuel residues (cladding hulls). Because of the type and quantity of material expected to be shipped in Type B packages there is concern about the ability of these containers to withstand accidents. Of primary concern is the possibility of a release in a very severe accident. Langhaar[7] reports that several Type B containers have been involved in severe accidents. No detectable radioactive release or excessive increase in radiation level has resulted.

Transportation accidents are relatively common. Since radioactive materials move in routine commerce, it is expected that they will sometimes be involved in accidents. Statistics show that about one package of radioactive material in 10,000 is involved in some kind of transport accident. However, the incidence of release of material is low. Most releases have involved low-specific-activity material or Type A packaging. Type B packaging, which is designed to withstand severe accident situations, has an excellent record of package integrity.

6.1.3 Transportation Accident Statistics

Transportation accidents have a wide range of severities. Most accidents occur at low vehicle speeds and have relatively minor consequences. In general, as speed increases, accident severity also increases. However, accident severity is not a function of vehicle speed only. Other factors such as the type of accident, the kind of equipment involved, and the location of the accident can have an important bearing on accident severity.

Furthermore, damage to a package in a transport accident is not directly related to accident severity. In a series of accidents of the same severity, or in a single accident involving a number of packages, damage to packages may vary from none to extensive. In relatively minor accidents, serious damage to packages can occur from impacts on sharp objects or from being struck by other cargo. Conversely, even in very severe accidents, damage to packages may be minimal.

Shipments of spent fuel and radioactive waste are assumed to be made via commercial carrier using conventional transportation equipment. These shipments would be subject to the same accident environment as commercial shipments of nonradioactive commodities. Accident data for truck shipments are compiled by the Bureau of Motor Carrier Safety of the Federal Highway Administration.(10) Accident data for rail shipments are compiled by the Office of Safety of the Federal Railroad Administration.(11) Summaries of hazardous materials accidents are compiled by the Office of Hazardous Materials of the U.S. Department of Transportation.

A recent study(12) has analyzed the probability of rail and truck accidents based on accident data supplied by the U.S. Department of Transportation. Accidents are classified by severity into five categories as functions of vehicle speed and fire duration. The five categories and their associated probabilities for both rail and truck accidents are shown in Table 6.1.3.

Consideration of the regulatory standards and requirements for package design and quality assurance, results of tests, and past experience support a conclusion that Type B packages will withstand all except very severe, highly unusual accidents. Such accidents are taken in the context of Table 6.1.3 to mean extra severe or extreme accidents. Even if an extra severe accident occurs package failure and subsequent release of radioactive contents do not necessarily occur but have only some probability. Estimates of the release probability for particular package designs have been discussed in References 13, 14, and 15.

In this study, accidents are grouped into four categories: minor, moderate, severe, and non-design basis. The definitions of these accident categories are given in Section 3.7. For transportation accidents analyzed in this study, minor accidents are assumed to have an accident probability of 1×10^{-6} per vehicle mile, moderate accidents are assumed to have an accident probability of 1×10^{-8} per vehicle mile, and severe accidents are assumed to have an accident probability of 1×10^{-11} per vehicle mile. Non-design basis accidents are assumed to have an accident probability less than 1×10^{-13} per vehicle mile and are not analyzed in this study.

6.1.4 Mitigating the Consequences of Transportation Accidents

In a transportation accident, carriers of radioactive material are required to follow DOT-prescribed procedures(16) designed to mitigate the consequences. DOT regulations require prompt reporting of any transportation incident involving shipment of radioactive material in

TABLE 6.1.3. Rail and Truck Accident Probabilities

Severity	Vehicle Speed, mph	Fire Duration, hr	Probability Per Vehicle Mile: Rail	Probability Per Vehicle Mile: Truck
Minor	0-30	<1/2	6×10^{-9}	6×10^{-9}
	0-30	0	4.7×10^{-7}	4×10^{-7}
	30-50	0	2.6×10^{-7}	9×10^{-7}
		Total	7.4×10^{-7}	1.3×10^{-6}
Moderate	0-30	1/2-1	9.3×10^{-10}	5×10^{-11}
	30-50	<1/2	3.3×10^{-9}	1×10^{-8}
	50-70	<1/2	9.9×10^{-10}	5×10^{-9}
	50-70	0	7.5×10^{-8}	3×10^{-7}
		Total	8.0×10^{-8}	3.1×10^{-7}
Severe	0-30	>1	7.0×10^{-11}	5×10^{-12}
	30-50	>1	3.9×10^{-11}	1×10^{-11}
	30-50	1/2-1	5.1×10^{-10}	1×10^{-10}
	50-70	1/2-1	1.5×10^{-10}	6×10^{-12}
	>70	<1/2	1×10^{-11}	1×10^{-10}
	>70	0	8×10^{-10}	8×10^{-9}
		Total	1.5×10^{-9}	8.2×10^{-9}
Extra severe	50-70	>1	1.1×10^{-11}	6×10^{-13}
	>70	1/2-1	1.6×10^{-12}	2×10^{-13}
		Total	1.3×10^{-11}	8×10^{-13}
Extreme	>70	>1	1.2×10^{-13}	2×10^{-14}
		Total	1.2×10^{-13}	2×10^{-14}

which fire, breakage, spillage, or suspected radioactive contamination occurs. The regulations also specify guidelines for remedial actions in situations involving actual or suspected release of radioactivity from a shipping container. Cars used for transporting radioactive material must be monitored after each shipment; they may not be returned to service until the dose rate on accessible surfaces is below prescribed levels and there is no significant removable radioactive surface contamination.

An intergovernmental radiological assistance program(17,18) provides personnel equipped to monitor radiation and trained to act as advisors to aid in radiological incidents such as a transportation accident involving nuclear material. The Federal radiological assistance program is coordinated by the Division of Operational Safety of DOE. It provides a mechanism whereby 13 Federal agencies coordinate their radiological emergency activities with the activities of state and local health, police, fire, and civil defense agencies.

In the event of a transportation accident, trained personnel from the radiological assistance program are available to help in the following activities:

- evaluate the radiological situation
- minimize personnel exposure to radiation and/or radioactive materials
- minimize the spread of radioactive contamination
- minimize damaging effects on property
- assist in carrying out emergency rescue and first aid procedures necessary to save life and minimize injury
- provide technical information to appropriate authorities and medical advice on the treatment of injuries complicated by radioactive contamination
- provide information to the public as quickly as possible in order to minimize undue public alarm and to assist in the orderly conduct of emergency activities.

REFERENCES FOR SECTIONS 6.0-6.1

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15. W. A. Brobst, "Transportation Accidents: How Probable?," Nuclear News, 16:(7) 48, May 1973.

16. Title 49, Code of Federal Regulations, Parts 171.15, 174.566, and 177.861 (49 CFR 171.15, 49 CFR 174.566, 49 CFR 177.861).

17. Interagency Radiological Assistance Plan, ERDA-10, Energy Research and Development Administration, Washington, DC, April 1975.

18. Energy Research and Development Administration Radiological Assistance Plan, ERDA-60, USERDA Division of Operational Safety, Washington, DC, July 1975.

6.2 TRANSPORTATION OF SPENT FUEL

6.2 TRANSPORTATION OF SPENT FUEL

For the "once-through" fuel cycle scenario, irradiated fuel elements are assumed to cool in a fuel storage basin at a nuclear reactor for a minimum of 6 months after discharge. Some of the fuel is then shipped by rail or truck to an interim storage facility. Some of the fuel remains in reactor basins for up to 6 years before shipment. After storage for about 6 years at a reactor basin or interim facility, the irradiated elements are overpacked in canisters at a packaging facility associated with one of the interim storage basins. They are then shipped by rail either to a Federally operated final isolation facility or to a Federally operated interim storage facility for packaged fuel.

Because of the presence of plutonium and other hazardous isotopes, irradiated nuclear fuel is classified as Transport Group I, the most restrictive grouping. Spent fuel elements constitute a "large quantity" of radioactive material under the definitions of 10 CFR 71. The primary reliance for safety in transport of irradiated nuclear fuel is based on shipping cask design. Spent fuel casks must satisfy the regulatory requirements for Type B packaging and must retain containment and shielding integrity if subjected to the hypothetical accident environment described in 10 CFR 71, Appendix B.

Spent fuel has been shipped in the United States for many years. Massive, heavily shielded shipping casks are available for both truck and rail transport of high-burnup fuel from current generation LWRs. Most spent fuel casks will accept either PWR or BWR spent fuel by using different fuel baskets; however, some are designed only for a particular fuel type (e.g., the TN series of Transnuclear, Inc.). Table 6.2.1 gives information about casks that are currently available or licensed for spent fuel shipments in the United States.

Spent fuel elements which are shipped by rail require criticality safety control during transportation. This control is obtained through the design of the casks used to transport the fuel. The considerable experience gained during the past 20 years demonstrates that criticality safety can be maintained for these shipments.

Currently available casks are designed to accept fuel assemblies without overpacks. Fuel shipped from interim storage to permanent isolation facilities is expected to be enclosed in canisters. The size of proposed canisters would necessitate some changes to existing cask designs. Casks could be modified to accept canistered fuel assemblies by increasing the length of a cask and by altering the fuel basket which is inserted into the cask cavity. Modified rail casks would accept fewer overpacked assemblies than bare assemblies.

The possibility of locating nuclear power plants offshore in lakes or estuaries or in the oceans a few miles from the nearest land is presently under investigation in the United States. The Nuclear Regulatory Commission has recently issued a draft environmental statement[1] for a proposed nuclear power station to be located in the Atlantic Ocean about 2.8 miles from the New Jersey shore. The movement of irradiated nuclear fuel and solid radioactive waste from offshore (floating) nuclear power plants will probably require transport by barge. Casks used for barge shipment of spent fuel would probably be like those used for rail shipment.

TABLE 6.2.1. Licensed and Available Shipping Casks for Current Generation LWR Spent Fuel

Cask Designation	Number of Assemblies PWR	Number of Assemblies BWR	Approximate Loaded Cask Weight, MT	Usual Transport Mode	Shielding Gamma	Shielding Neutron	Cavity Coolant	Maximum Heat Removal, kW	Status
NFS-4 (NAC-1)	1	2	23	Truck	Lead and steel	Borated water and antifreeze	Water	11.5	6 casks available
NLI 1/2	1	2	22	Truck	Lead, uranium and steel	Water	Helium	10.6	3 casks available
TN-8	3		36	Truck[(a)]	Lead and steel	Borated solid resin	Air	35.5	Licensed
TN-9		7	36	Truck[(a)]	Lead and steel	Borated solid resin	Air	24.5	Licensed
IF-300	7	18	63	Rail[(b)]	Uranium and steel	Water and antifreeze	Water	76[(c)]	4 casks available
NLI 10/24	10	24	88	Rail	Lead and steel	Water	Helium	97[(d)]	Licensed

a. Overweight permit required.
b. Truck shipment for short distances with overweight permit.
c. Licensed decay heat load is 62 kW.
d. Licensed decay heat load is 70 kW.

6.2.1 Rail Transport of Spent Fuel

Several factors can influence the choice of rail or truck casks for the shipment of spent fuel. Rail casks have a significantly larger payload capacity than truck casks. (The payload capacity of currently licensed rail casks is about 52 kg of heavy metal per metric ton of cask weight whereas the payload capacity of truck casks is only about 18 kg of heavy metal per metric ton of cask weight.) On the other hand, shorter truck travel times tend to compensate for the payload to gross weight advantage of rail shipments. The time required to load or unload a rail cask is about twice as long as that required to load or unload a truck cask. However, considering that a rail cask can transport up to ten times as much spent fuel as a truck cask, the loading or unloading time per ton of fuel is lowest for rail shipment.

It is assumed that rail casks will be used for spent fuel shipment whenever practicable. However, of the 89 reactors scheduled for operation by 1980, only 50 reactors (56%) have rail spurs at the site.[2] It is probable that some reactors without rail spurs will be serviced by intermodal casks that are shipped short distances by truck with overweight permits. This allows the cask to be used to transport fuel from reactors without a rail siding to the nearest railhead where the shipment is completed by rail.

It is assumed that 90% of spent fuel will be shipped by rail and 10% by truck. (This same assumption was made in an analysis of the risk of transporting spent fuel recently published by the Environmental Protection Agency.)[3] Shipping information for spent fuel shipments from nuclear reactors to interim storage and packaging facilities is shown in Table 6.2.2 based on an installed capacity of 400 GWe in the year 2000. Table 6.2.3 gives year 2000 shipping information for canistered spent fuel shipped from interim storage and packaging to a Federally operated final isolation facility. It is assumed that canistered spent fuel would be shipped in the modified rail cask described in Section 6.2.1.1

6.2.1.1 Rail Casks for Shipment of Spent Fuel

As indicated in Table 6.2.1, two rail casks, the IF-300 and the NLI 10/24, are currently licensed for rail shipment of spent fuel in the United States.

IF-300

The IF-300 cask (Figure 6.2.1) of General Electric Company is a water-filled cask designed for rail transport of 7 PWR or 18 BWR fuel assemblies.[4,5] Approximate loaded cask weight is 63 MT (140,000 lb). The gross weight of one cask, its skid, and auxiliary components is 80 MT (175,000 lb). The cask is normally transported by rail on a 100-MT capacity, four-axle flatcar, but it may be transported for short distances on highways on a special nine-axle truck requiring special overweight permits. This special truck is used to move the cask from reactors without rail facilities to the nearest rail siding. Four IF-300 casks have been built.

The cask has an overall length of 5.33 m (210 in.) and a diameter of 1.62 m (64 in.). The cavity has a length of 4.58 m (180 1/4 in.) and a diameter of 0.95 m (37 1/2 in.). Interchangeable fuel baskets provide it with a capacity of 7 PWR of 18 BWR fuel assemblies.

Gamma shielding is provided by 10 cm (4 in.) of depleted uranium sandwiched between stainless steel inner and outer shells. Neutron shielding is furnished by the water in the

TABLE 6.2.2. Shipping Information for Irradiated Fuel Assemblies Shipped from Reactors to Interim Storage Basins and Packaging in the Year 2000

	PWR	BWR
Installed capacity[a]	267 MWe	133 MWe
Total weight of fuel shipped per yr[b]	3852 MTHM	2411 MTHM
Total number of assemblies shipped per yr	8350	12,780
Fuel plus fission products per assembly	461.4 kg	188.7 kg
Assemblies per cask		
NLI 10/24	10	24
IF-300	7	18
Truck cask	1	2
Shipments per yr[c]		
NLI 10/24	376	240
IF-300	537	320
Truck cask	835	639
Assumed shipping distance, one way	1000 miles	1000 miles
Shipping time, one way		
Rail cask	7 days	7 days
Truck cask	1.5 day	1.5 day
Radioactivity per cask[d]		
NLI 10/24	2.2×10^6 Ci	1.6×10^6 Ci
IF-300	1.5×10^6 Ci	1.2×10^6 Ci
Truck cask	2.2×10^5 Ci	1.3×10^5 Ci
Thermal power per cask[d]		
NLI 10/24	7.1 kW	5.8 kW
IF-300	5.0 kW	3.7 kW
Truck cask	0.07 kW	0.4 kW

a. Total installed capacity is 400 GWe. Reactors are assumed to operate 70% of the time to produce 280 GWe/yr.
b. Based on 62% PWR and 38% BWR fuel by weight.
c. Based on an assumed mix of 45% of fuel assemblies shipped by NLI 10/24, 45% shipped by IF-300, and 10% shipped by truck casks.
d. Assumes 6 years cooling prior to shipment.

TABLE 6.2.3. Shipping Information for Canistered Fuel Assemblies Shipped from Interim Storage and Packaging to a Federally Operated Final Isolation Facility in the Year 2000[a]

	PWR	BWR
Total weight of fuel shipped per yr	3402 MTHM	2139 MTHM
Total number of assemblies shipped per year	7370	11,340
Number of assemblies per cask	7	17
Number of shipments per yr	1054	667
Assumed shipping distance, one way	1500 miles	1500 miles
Shipping time, one way	10 days	10 days
Radioactivity per cask[b]	1.5×10^6 Ci	1.1×10^6 Ci
Thermal power per cask[b]	5 kW	3.5 kW

a. Shipment is assumed to be made by rail in a cask modified as described in Section 6.2.1.1.
b. Assumes 6 yr cooling prior to shipment.

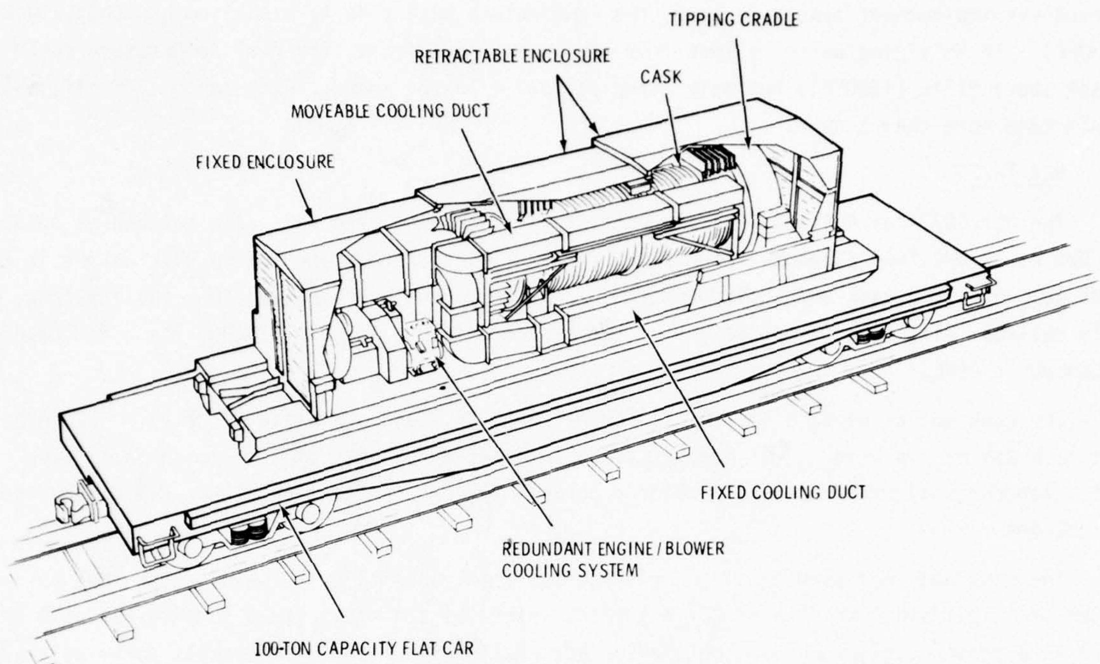

FIGURE 6.2.1. IF-300 Shipping Cask Shown in Normal Rail Transport Configuration

cask cavity plus 11.4 cm (4 1/2 in.) of water contained in a stainless steel corrugated jacket. Freezing in the winter is prevented by using a mixture of water and ethylene glycol. Criticality control is provided by boron carbide-filled stainless steel tubes that are welded to the fuel baskets. Impact protection is provided by stainless steel fins mounted radially and on the cask heads. The external water jacket, made of thin-walled material, does not contribute to the impact protection of the cask.

A single cask lid is made of stainless steel, forged plates. Thirty-two 1-3/4-in.-dia stud bolts attach the lid. The closure head is sealed with a Grayloc metallic ring. The cask cavity has two bellows sealed and stainless steel globe valves for filling, draining, and sampling. To prevent loosening, the valve handles are lockwired during transit.

The cavity maximum normal operating pressure is 200 psig. However, the design working pressure is 400 psig at a material temperature of 324°C (615°F). A combination breaking pin and pressure relief valve protect the cavity from overpressure. Discharge pressure for the pin and valve is 350 psig. The discharge valve is set for a maximum steam or gas blowdown of 5% and a liquid blowdown of 10%.

Heat is removed from the fuel to the cask cavity walls by natural circulation of the contained water, by conduction through the cask walls to the outer neutron shield, and by convection through the neutron shield to the corrugated outer wall. Two diesel-driven blowers and appropriate air ducts facilitate external cooling of the corrugated wall and steel fins. Maximum heat rejection capacity is 76 kW with blowers operating and 62 kW without the blowers. During normal operation, the maximum fuel temperature is expected to be 163°C (326°F). If the

forced air impingement system is lost, the temperature will rise to a maximum of about 220°C (430°F). If shielding water is lost from the outer compartment, the fuel temperature could reach about 815°C (1500°F), but only after all water in the inner cavity had boiled off, which would take more than 2 days.

NLI 10/24

The NLI 10/24 of National Lead Industries is a helium-filled rail cask capable of holding 10 PWR or 24 BWR fuel elements (Figure 6.2.2).[6,7] The approximate loaded cask weight is 88 MT (193,000 lb). The cask and cooling systems are transported on a special 18-m (59-ft) long, six-axle railroad flat car. Total weight of the system is about 152 MT (335,000 lb). The cask was licensed in 1976.

The cask has an overall length of 5.19 m (204.5 in.) and a diameter of 2.24 m (88 in.). The cask cavity has a length of 4.56 m (179.5 in.) and a diameter of 1.14 m (45 in.). Two interchangeable aluminum baskets provide a capability for transporting either PWR or BWR fuel assemblies.

The cask body consists of an inner stainless steel shell 2 cm (3/4-in.) thick and an outer stainless steel shell 5 cm (2-in.) thick joined by stainless steel forgings at each end to make a continuous weldment. The annulus between the inner and outer shells contains a lead gamma shield 15 cm (6-in.) thick. Depleted uranium shielding is used on the ends of the cask and at strategic locations in the wall of the cask. Neutron shielding is provided by 23 cm (9 in.) of water contained in a finned stainless steel jacket surrounding the outer shell. Criticality control is provided by the stainless steel clad Ag-In-Cd liners of the aluminum fuel baskets. Balsa impact limiters at each end of the cask, in addition to the circumferential cooling fins, give impact protection.

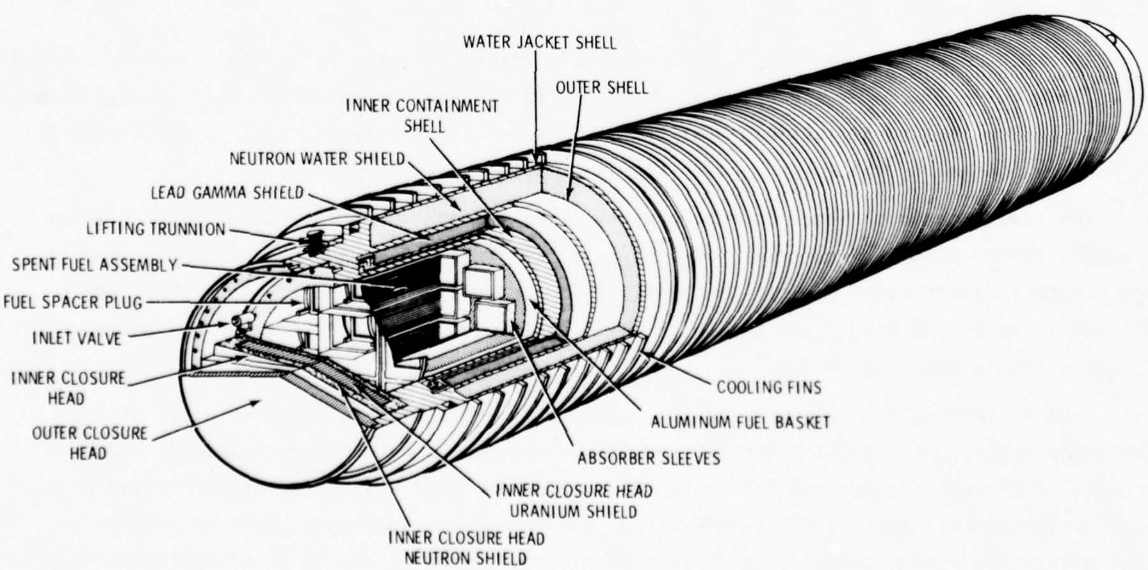

FIGURE 6.2.2. NLI 10/24 Rail Cask Assembly

Two closure heads are used for the cask. The outer closure head is a stainless steel plate 6.4 cm (2-1/2 in.) thick held in place by twenty-four 1-1/4 in. high-strength studs. The inner closure head is a stainless steel forging filled with depleted uranium and sealed with a metallic ring. The inner closure head is held in place by twenty 1-1/2 in. high-strength studs. One penetration through the head end forging of the cask body exits into the space between the inner and outer closure heads. This is used to drain this space and to pressure test the secondary containment system prior to shipment. The four penetrations in the inner closure head cask flange are used for servicing the cask cavity. They are equipped with globe-type angle valves and terminate in the secondary containment volume between the inner and outer heads.

Maximum decay heat load for the NLI 10/24 is 97 kW. The aluminum basket serves as a heat conduction path from each fuel assembly to the cavity wall. Decay heat is removed from the cavity wall by means of cooling water circulated through channels welded to the outside surface of this wall. The cooling channels terminate in two separate header arrangements connected to separate heat exchanger systems. Without auxiliary cooling and at maximum heat generation rate, the average fuel temperature is 348°C (659°F). Without the cooling system in operation, heat dissipation is by conduction through the cask body to the neutron shield, convection to the outer surface, and convection plus radiation from the finned outer surface to the atmosphere. Maximum fuel temperature during fire accident conditions is 533°C (991°F). Normal cavity pressure during transport is expected to be about 23 psig, with a maximum internal pressure of 105 psig occurring in the fire accident. The NLI 10/24 containment vessel has a maximum allowable working pressure several times greater than this value.

Shipment of spent fuel from interim storage to permanent isolation will require use of rail casks designed to accept canistered fuel assemblies. Existing rail casks would have to be about 0.6 m (2 ft) longer to accommodate canistered assemblies. Different fuel baskets would also be required because of the larger canister cross section. It is assumed that a suitably modified rail cask based on the NLI 10/24 design could accommodate 7 PWR or 17 BWR canistered assemblies.

6.2.1.2 Secondary Wastes from Rail Shipment of Spent Fuel

Secondary wastes are generated in cask loading and unloading operations at nuclear reactors. interim storage facilities, and at the Federal facility for final isolation. Generation and treatment of these wastes is discussed in the receiving and shipping facility descriptions in other sections of this document.

6.2.1.3 Emissions from Rail Shipment of Spent Fuel

Shipments of spent fuel and radioactive waste are subject to radiation dose rate limitations prescribed by the U.S. Department of Transportation (see Section 6.1.1). For shipments in closed vehicles, Federal regulations impose dose rate restrictions of 200 mrem/hr at the external surface of the vehicle, 10 mrem/hr at any point 1.8 m (6 ft) from the vehicle, and 2 mrem/hr at any normally occupied position in the vehicle. Actual experience in transporting spent fuel[8] has shown that radiation levels from cask shipments rarely exceed 60 mrem/hr at the

vehicle surface or 0.2 mrem/hr in the truck cab. Reference 8 also analyzes the radiation levels to which transportation workers and the general public would be expected to be exposed as a result of routine shipments of spent fuel.

The rate of heat release to the air from spent fuel casks varies from about 10 kW for truck casks to about 100 kW for large rail casks. These heat release rates can be compared to the rate at which waste heat is released from a 100-hp truck engine operating at full power, which is about 50 kW. Federal regulations limit the temperature of the accessible surface of a shipping container to 82°C (180°F). Hence the temperature of the air that contacts a cask shipment will be increased a few degrees, but the temperature of the air a few feet from the shipment will be unaffected.

6.2.1.4 Decommissioning of Spent Fuel Rail Casks

The useful life of a spent fuel cask is estimated to be 20 to 30 years. Decommissioning is assumed to be accomplished by appropriate decontamination procedures followed by disposal as non-TRU waste.

6.2.1.5 Postulated Accidents for Rail Shipment of Spent Fuel

The key design goal of spent fuel casks is to provide safe transport of spent fuel (i.e. no radiation leakage). The casks are rugged, thick-walled containers designed to retain shielding and containment integrity in virtually all transport accident situations. For licensing these casks, a set of hypothetical accident conditions (Type B package tests) have been specified which simulate severe accident environments. Type B test criteria include impact, puncture, fire and immersion tests. Recently, the protection provided by Type B package requirements has been investigated by intensive analytical studies,[9-11] by small-scale and model studies,[12-15] and by full-scale tests of actual shipping containers.[9,16,17] Additional studies and tests are planned. Results of cask test programs indicate that spent fuel casks will withstand all but very severe, highly unusual accidents.

A spent fuel cask provides double containment for irradiated fuel during transport. The first barrier to an accidental release of radioactivity is the fuel cladding. The second barrier is the combination of inner shell, gamma shield, and outer shell which constitute the cask body. For a release to occur during transport, radioactive material must first escape to the cask cavity from perforated fuel rods. The material must then be released from the cask atmosphere as a result of a penetration of cask containment. The cask may leak by one of three paths: actuation of pressure relief mechanisms, loss of closure head seal, or breach of the cask. Because of the cask construction, barrier failure from accidental breach of the cask is considered extremely unlikely.

Most cask designs include a pressure relief system, expected to vent when the internal pressure exceeds a preset level. Most systems are designed to reseal after excess pressure is relieved.

A breach of containment resulting from a transport accident would probably consist of cracks in the fuel rod cladding coupled with pressure relief or cask head closure failure. It would release only a small fraction of the radioactivity contained in a cask. The most

likely result of an accident that breaches a cask would be release of some cavity coolant (water or helium). Radioactivity in the cavity coolant would also be released. Very severe accidents might result in the rupture of some fuel rods and a release of fission gases from fuel rod void spaces. A high-temperature environment severe enough to perforate fuel rods would probably cause release of a small fraction of the cesium contained in these rods.

Tables 6.2.4, 6.2.5, and 6.2.6 describe postulated minor, moderate, and severe accident scenarios for rail shipment of spent fuel from reactors to interim spent fuel storage facilities. Although several transportation accidents involving spent fuel casks and other Type B packages have occurred, no transportation accident involving a Type B container has actually resulted in a release of radioactivity to the environment.

Expected frequencies for accidents postulated in Tables 6.2.4, 6.2.5, and 6.2.6 are based on accident probabilities per vehicle mile from Section 6.1.3 and on total shipment miles calculated from data in Table 6.2.2.

For purposes of environmental consequence analysis, the material releases associated with accident numbers 6.2.3, 6.2.6 and 6.2.8 in Tables 6.3.4, 6.2.5 and 6.2.6 have been selected as umbrella source terms. (The concept of an umbrella source term is explained in Section 3.7.) This means that the releases from these accidents are the largest in their respective source term categories. The environmental consequences of these accidents are described in DOE/ET/0029. Accidents are cross indexed with their appropriate umbrella source term in Appendix A, Section 3.

Neutron emission rates shown in Table 6.2.5 are based on the spontaneous fission of ^{242}Cm and ^{244}Cm and on (α, n) reactions with oxygen. Neutron emission rates for curium compounds[18] are calculated to be 4.36×10^7 n/sec-g for $^{242}CmO_2$ and 1.10×10^7 n/sec-g for $^{244}CmO_2$.

A fraction of the volatile fission products in a spent fuel element is released from the fuel matrix and accumulates in rod void spaces. Table 6.2.7 provides an estimate[19] of fission gas fractions that accumulate in fuel rod void spaces. If the fuel cladding is ruptured, these fission gases could escape. Because of the regulatory limits in 10 CFR 71.35 on the radioactivity in the cask coolant, any fuel assembly that is releasing a significant amount of radioactivity must be placed in a separate, sealed container (i.e., "canned") prior to being loaded into a cask for shipment. Fuel assemblies releasing significant amounts of radioactivity while in a reactor will have been identified before being discharged from the reactor, but some so-called "failed fuel" may go undetected. In the case of "failed fuel", much of the radioactivity in the fuel rod void spaces may have been released while the assembly was still in the reactor after failure or while the assembly was stored in the reactor storage basin prior to shipment.

It is believed conservative to assume that, under normal conditions of transport, 0.25% of the free gases and other activities from fuel rod void spaces would be outside the fuel assemblies in the cask coolant during shipment of the fuel assemblies.[8] The cask coolant might also be contaminated from fuel rod surface contamination and from residual contamination from

TABLE 6.2.4. Spent Fuel Rail Cask Minor Accidents

Accident No. and Description	Sequence of Events	Safety System	Release
6.2.1 – Train derailment involves spent fuel cask Expected frequency $\sim$2 per year.	1. Derailment accident occurs. 2. Railcar leaves track and may overturn. 3. Confinement barriers of cask remain intact. 4. Accident is reported to local and Federal officials. 5. Railroad crews restore car to track.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site and recover cask. 3. Confinement barriers of cask contain all radioactive materials.	None
6.2.2 – Train derailment 1/2 hr (or less) fire involves spent fuel cask. Expected frequency $\sim$0.2 per year.	1. Derailment accident occurs. 2. Railcar leaves track and may overturn. 3. Spent fuel cask is involved in 1/2 hr (or less) fire. 4. Confinement barriers of cask remain intact. 5. Accident is reported to local and Federal officials. 6. Railroad crews restore car to track.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site and recover cask. 3. Confinement barriers of cask contain all radioactive materials.	None
6.2.3 – Undetected leakage of coolant from cask cavity (and/or surface contamination washoff). Expected frequency $\sim$2 per year.	1. A defective gasket or valve results in a leak of not more than 0.001 cc/sec (below the detection threshold). 2. Leakage of cavity coolant occurs for the duration of the trip. 3. Cavity coolant contains some radioactivity due to transportation with a small percentage of fuel rods with failed cladding.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Checks made on cask prior to release for shipment would detect larger leaks.	For casks which use water as a cavity coolant, assume 1 Ci of gross fission product activity in coolant (1 μ Ci/cc). In 4 days an undetected leak of 0.001 cc/sec would release about 350 μCi of activity. Estimate 1% of the released activity in the liquid is dispersed in the form of an aerosol.

TABLE 6.2.5. Spent Fuel Rail Cask Moderate Accidents

Accident No. and Description	Sequence of Events	Safety System	Release
6.2.4 — Collision or derailment results in loss of neutron shielding from spent fuel rail cask. Expected frequency $\sim 2\times10^{-2}$ per year.	1. Collision and/or derailment accident occurs at moderate speed. 2. Railcar overturns. 3. Cask leaves railcar. 4. Water jacket is ruptured and neutron shield water is lost. 5. Accident is reported to local and Federal officials. 6. Cask recovered.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site, minimize personnel exposure, and recover cask. 3. Remaining confinement barriers of cask remain intact and contain all radioactive materials.	No radioactive material release Neutron emission rates are: 4.36×10^7 n/sec-g $^{242}CmO_2$ 1.10×10^7 n/sec-g $^{244}CmO_2$. Mean neutron energy is 2 MeV.
6.2.5 — Fire accompanying accident causes a rail cask cavity to overpressurize; relief valve operates. Expected frequency $\sim 2\times10^{-2}$ per year.	1. Collision and/or derailment accident occurs at moderate speed. 2. Spent fuel cask involved in 1/2 hr (or longer) fire. 3. Cask cavity overpressurizes and pressure relief valve operates to relieve pressure. 4. Mechanical cooling system remains operational. 5. 0.1% of cavity coolant is released from cask due to operation of pressure relief valve. 6. Cavity coolant contains some radioactivity due to transportation with a small percentage of fuel rods with failed cladding. 7. Accident is reported to local and Federal officials. 8. Area decontaminated. 9. Cask recovered.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site, minimize personnel exposure, and recover cask. 3. Confinement barriers of cask remain intact and limit release to a small percentage of cavity coolant.	Venting of cask occurs at ground level without off-gas system control. Assuming 0.25% of fuel rods exhibit failed cladding, the following release fractions apply: Isotope — Release Fraction ^{85}Kr — 7.5×10^{-7} ^{129}I — 2.5×10^{-7} ^{3}H — 2.5×10^{-7} For casks which use water as cavity coolant, assume an additional 1 Ci of gross fission product activity in the coolant. Venting of "wet" cask results in release of 1×10^{-3} Ci of gross fission products. Release fractions apply to cask inventory of 4 MTHM. Use release period of 1 hr. Tables 3.3.8 and 3.3.10 describe activity spectrum of the 0.5-year-old fuel.

TABLE 6.2.5. *(contd)*

Accident No. and Description	Sequence of Events	Safety System	Release
6.2.6 — Collision or derailment causes damage to rail cask mechanical cooling system. Expected frequency $\sim 2 \times 10^{-2}$ per year.	1. Collision or derailment accident occurs at moderate speed. 2. Railcar overturns. 3. Spent fuel cask may be involved in 1/2 hr (or longer) fire, or ambient temperature is above 38°C (100°F). 4. Mechanical cooling system becomes temporarily inoperative. 5. Cask cavity overpressurizes. 6. Venting of cask cavity occurs in a series of releases resulting in a loss of 5% of the cavity coolant. 7. Accident is reported to local and Federal officials. 8. Area decontaminated. 9. Cask recovered.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site, minimize personnel exposure, and recover cask. 3. Fuel rods do not perforate and there is no gross breach of cask containment.	Venting of cask occurs at ground level without off-gas system control. Assuming 0.25% of fuel rods exhibit failed cladding, the following release fractions apply: Isotope — Release Fraction ^{85}Kr — 4×10^{-5} ^{129}I — 1×10^{-5} ^{3}H — 1×10^{-5} For casks which use water as cavity coolant, assume an additional 1 Ci of gross fission product activity in the coolant. Venting of "wet" cask results in release of 5×10^{-2} Ci of gross fission products. Release fractions apply to cask inventory of 4 MTHM. Use release period of 1 hr. Tables 3.3.8 and 3.3.10 describe the activity spectrum of 0.5-year-old fuel.

TABLE 6.2.6. Spent Fuel Rail Cask Severe Accidents

Accident No. and Description	Sequence of Events	Safety System	Release
6.2.7 – Collision or derailment causes spent fuel rail cask to be subjected to severe impact and fire. Expected frequency $\sim 2 \times 10^{-5}$ per year.	1. Collision and/or derailment accident occurs at high speed. 2. Spent fuel cask strikes massive object and decelerates almost instantaneously. 3. Cask is involved in a fire which lasts longer than 1 hour. 4. Closure head seal breached by gasket failure or by cask lid bolts being sheared off. 5. 1% of fuel rods are failed and 100% of cavity coolant is released. 6. Accident is reported to local and Federal officials. 7. Cask sealed. 8. Area decontaminated. 9. Cask recovered.	1. Interagency radiological assistance personnel available to assist local public safety and carrier officials to control site, minimize personnel exposure, and recover cask. 2. Only a small opening exists in cask. Accident does not result in gross breach of cask containment.	Ground level release occurs with following release fractions: Isotope — Release Fraction ^{85}Kr — 3×10^{-3} ^{129}I — 1×10^{-3} ^{3}H — 1×10^{-3} Mixed fission products — 1×10^{-8} Actinides — 1×10^{-8} Release fractions apply to cask inventory of 4 MTHM. Use release period of 2 hours. Use Tables 3.3.8 and 3.3.10 for activity spectrum of 0.5-year-old fuel.
6.2.8 – Cavity coolant lost from spent fuel rail cask; no emergency action taken. Expected frequency $\sim 2 \times 10^{-5}$ per year.	1. No emergency action taken to cool spent fuel cask of accident number 6.2.6 for a period of at least 10 hours. 2. Ambient temperature is above 38°C (100°F). 3. Increase in cavity pressure causes rupture disk to fail. 4. 100% of cavity coolant is lost. 5. Cladding failure occurs in 50% of fuel rods when rod temperatures reach ∿650°C (1200°F). 6. Fission gases and some cesium released from perforated rods. 7. Accident is reported to local and Federal officials. 8. Cask sealed. 9. Area decontaminated.	1. Interagency radiological assistance personnel available to assist local public safety and carrier officials to control site, minimize personnel exposure, and recover cask. 2. Only a small opening exists in cask. Accident does not result in gross breach of cask containment.	Ground level release occurs with following release fractions: Isotope — Release Fraction ^{85}Kr — 1.5×10^{-1} ^{129}I — 5×10^{-2} ^{3}H — 5×10^{-2} ^{134}Cs — 3×10^{-4} ^{137}Cs — 3×10^{-4} Release fractions apply to cask inventory of 4 MTHM. Use release period of 6 hr. Use Tables 3.3.8 and 3.3.10 for activity spectrum of 0.5-year-old fuel.

TABLE 6.2.7. Estimated Activity Available for Release in Penetration of Fuel Rod Cladding[19]

Radioactive Material	Percent in Fuel Rod Void Spaces
^{85}Kr	30
^{129}I	10
^{3}H	10

the storage pool since the cask loading operation is carried out in the storage pool water. A level of 1 μCi/cc has been estimated for this coolant contamination.[8] The activity includes a mixture of activation, corrosion and fission products.

Current spent fuel rail casks employ auxiliary cooling systems to aid in removal of internally generated heat. If the auxiliary cooling system were to become inoperative as the result of a severe transportation accident, and no corrective action is taken to cool the cask, it is conceivable that a total loss of cavity coolant might occur. The cavity coolant serves normally as the transfer medium for heat generated in spent fuel elements by the radioactive decay of fission products. If emergency action is not taken to cool the spent fuel following an accident in which cavity coolant is lost, the temperature of the fuel rod may rise significantly. For one rail cask design it has been calculated[5,8] that a loss of cavity coolant could result in an equilibrium fuel pin temperature of 858°C (1576°F) being attained after several hours. Experiments[5] have shown that fuel rods will perforate as a result of several hours' exposure to a temperature above 650°C (1200°F), releasing all the volatile fission products in fuel rod void spaces and a fraction of the cesium. Metallic cesium has a boiling point of 670°C (1238°F). A recent NRC study has been made[20] of cesium releases from irradiated fuel in a transportation accident. This study estimates that 6×10^{-4} of the cesium inventory in a spent fuel rod may be in the form of metallic cesium which has migrated to void boundaries and may be available for release as a result of fuel rod perforation in a high-temperature environment.

Impact and puncture environments specified for Type B test criteria are unlikely to be exceeded in any but the most severe transportation accidents. Even these severe accidents would probably result in very small penetrations of the cask, which would limit the release of radioactivity to the environment. Specific data on fines production from fuel rod fracture are not available. Data are available[21] on respirable fines production from high-speed impacts of stainless steel canisters containing borosilicate glass similar to that proposed for high-level waste solidification. On the basis of "worst-case" data from Reference 21, it is assumed that a fuel rod fracture results in 10^{-3} of the fuel being converted to sub-10 μ particles. Only a small fraction of the respirable fines produced in a severe accident would escape to the cask cavity from cracks in a fuel rod and from the cask cavity to the environment through small penetrations of cask containment.

6.2.1.6 Cost of Rail Transport of Spent Fuel

For the conceptual system assumed in this analysis, rail or truck transport of spent fuel as a waste product in the once-through cycle will be required from the reactor to an independent spent fuel storage and packaging facility. Rail transport will be required for the packaged fuel from this facility to the Federal repository or to an interim storage facility. The rail shipment requirements assumed for the purposes of developing transport costs are illustrated schematically in Figure 6.2.3. Estimates, in mid-1976 dollars, of capital, freight, and total unit costs for both of the rail shipments are developed in this section.

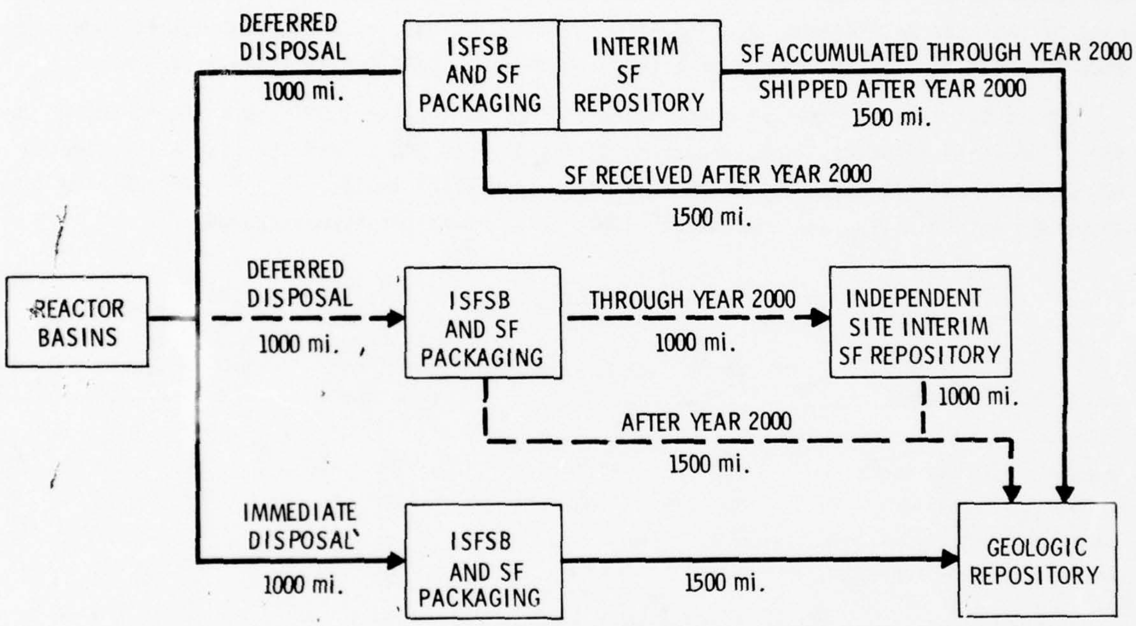

FIGURE 6.2.3. Spent Fuel Shipping Alternatives

Capital Cost of Spent Fuel Rail Cask System

The capital cost of the waste transportation system for rail shipment of spent fuel is estimated to be $3 million, with an accuracy range of ±40%. In making this estimate the system is treated as purchased equipment supplied repetitively by qualified vendors on a competitive basis. The estimate reflects manufacturer's profits and development, engineering, sales, overhead, and other similar expenses in addition to material and manufacturing costs.

The capital cost estimate covers costs for the complete transportation system, including the cost of the cask, rail car, tiedown system, auxiliary cooling equipment, and sun shield. Items excluded from the estimate are the owner's technical, procurement, and general administration expenses associated with the purchase of the transport system.

The accuracy range reflects the uncertainties in the optimum design of a spent fuel rail cask. Additionally, the range reflects the spread in pricing normally experienced in quotes from manufacturers of specialized heavy equipment.

Cask-Use Charge for Spent Fuel Rail Cask

A cask use charge was calculated for a spent fuel rail cask using the capital cost above and the levelized cost procedures outlined in Section 3.8 (assuming private ownership). A cask use factor of 80% or 292 days per year and an annual maintenance charge of 2% of the initial capital cost were also assumed. On this basis a daily use charge of about $2700/day was calculated. The cask use factor allows for time when the cask is not available for service--cask or cask car maintenance, routine licensing inspections, etc. It does not include allowance for idle time at either end of a trip, which is allowed for as a separate item.

Round-trip travel times were estimated at 13 days for the 1000-mile shipment and 20 days for a 1500-mile shipment. Total turnaround time to load and unload the cask was estimated at 4 days. The total cask-day calculations are summarized in Table 6.2.8. Multiplying total cask-days by the daily cask use charge gives the round-trip cask use charge.

TABLE 6.2.8. Calculation of Cask-Days Per Trip--Regular Train

Waste Shipment	Distance, miles	Casks Carried/ Trip	Round-Trip Travel Time/ Cask, days	Round-Trip Turnaround Time/ Cask, days	Cask-Days/ Trip
Spent fuel - rail					
Reactor → ISFS and packaging	1000	1	13	4	17
Packaging → repository	1500	1	20	4	24
Packaging → interim storage	1000	1	13	4	17
Interim storage → repository	1000	1	13	4	17

Unit Cost Estimate for Rail Transport of Spent Fuel

The round-trip freight charge is based on freight charges calculated for spent fuel shipments in Reference 22. Figure 6.2.4 shows a plot of data from this document. A regression analysis of the data resulted in the equation shown in the figure. This equation was used to calculate the dollar freight charge.

The unit transport cost was derived by adding the round-trip freight charge to the previously calculated round-trip cask use charge as shown in Table 6.2.9. In Table 6.2.9 each charge is expressed in terms of dollars per kilogram of fuel capacity in each cask. Since cask capacities vary with fuel type and cask type, the unit cask-use charge also varies.

Special Train Alternative

There are currently proposals by railroads to require shipment of spent fuel and nuclear waste in special trains. These trains would travel no faster than 35 mph, be required to stand still whenever another train was passing, and would carry no other cargo. The economic effect of a special train requirement was briefly studied in this document as an alternative to the regular train base case.

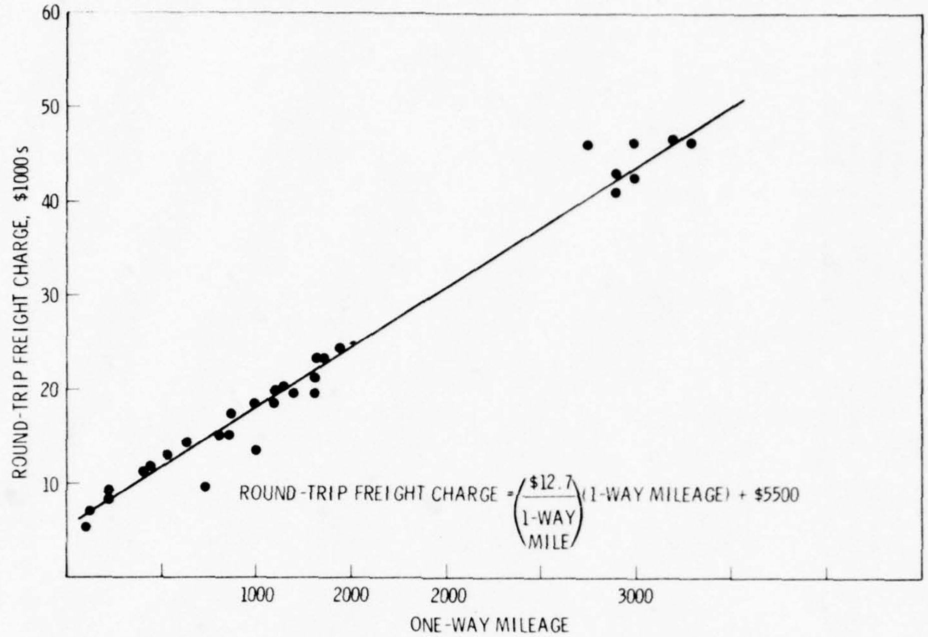

FIGURE 6.2.4. Estimated Freight Charges for Rail Cask Transport[22]

Reference 22 indicates an average special train charge of about $19/mile. Because of its impact on cask use charges, the average transport time for a special train is a critical cost parameter. If special trains travel faster than regular freight because, for example, they do not spend as much time in switch yards, then reduced use charges would tend to offset the special train charge. If, on the other hand, special trains travel more slowly because of the limitation on maximum speed, then use charges would increase and the cost of special train transport would be even greater than just the addition of the special train charges. There is much uncertainty concerning the probable average transport time of special trains. For this study we assumed special trains would travel twice as fast as regular trains. Special train charges can be minimized by shipping multiple casks per train. However, this may increase the round-trip cask use charges if casks must wait while other casks are being loaded or unloaded. The optimal number of casks per train is a function of the daily cask use charge, round-trip turn-around time, and special train charge. For the reference spent fuel casks and shipping distances up to 1500 miles, one cask per train is estimated to be optimal. The cost of the special train alternative under the above conditions is shown in the far right column of Table 6.2.9.

6.2.1.7 Construction Requirements for Spent Fuel Rail Casks

An estimate has been made of the approximate quantities of basic construction materials employed in fabricating a spent fuel rail cask. The estimate is given in Table 6.2.10 and represents average requirements for the rail casks shown in Table 6.2.1. Quantities listed in the table do not include allowances for canisters, rail cars, support systems, or other appurtenances.

TABLE 6.2.9. Unit Cost Estimate for Rail Transport of Spent Fuel (Regular Train)

Waste Type and Destination	Cask Type	One-Way Shipping Distance, miles	Cask Capacity, MTHM	Casks/Trip	Daily Cask Use Charge, $/MTHM/day	Cask-Days/Trip	Total Cask-Use Charges, $/kg HM	Round-Trip Freight Charge, $/kg HM	Total Unit Cost,[b] $/kg HM	Unit Cost Assuming Special Train Requirement, $/kg HM
Spent fuel to interim storage and/or packaging										
PWR	NLI 10/24	1000	4.61	1	575	17	9.78	4.00	13.80	18.00
	IF-300	1000	3.23	1	820	17	13.94	5.65	19.60	25.60
BWR	NLI 10/24	1000	4.53	1	585	17	9.95	4.00	14.00	18.30
	IF-300	1000	3.40	1	780	17	13.26	5.45	18.70	24.40
Packaged spent fuel from interim storage to geologic isolation										
PWR	Modified NLI 10/24	1500	3.23	1	822[a]	24	19.70	7.60	27.30	36.70
BWR	Modified NLI 10/24	1500	3.20	1	830[a]	24	19.90	7.70	27.60	37.10
Packaged spent fuel to independent Federal interim storage										
PWR	Modified NLI 10/24	1000	3.23	1	822	17	14.00	5.60	19.60	25.60
BWR	Modified NLI 10/24	1000	3.20	1	830	17	14.10	5.70	19.80	25.90
Packaged spent fuel from independent Federal interim storage to geologic isolation										
PWR	Modified NLI 10/24	1000	3.23	1	822	17	14.00	5.60	19.60	25.60
BWR	Modified NLI 10/24	1000	3.20	1	830	17	14.10	5.70	19.80	25.90

a. Assumes that the modified cask for shipping the packaged spent fuel will cost the same as the reference spent fuel cask.
b. The uncertainty in the unit cost estimate is ±50%.

TABLE 6.2.10. Construction Materials for Spent Fuel Rail Cask

Material	Quantity
Steel	26,000 kg
Lead	65,000 kg
Depleted uranium	5,000 kg

6.2.1.8 Effects of Fuel Cycles on Rail Shipment of Spent Fuel

The reference process for rail shipment of spent fuel assumes no recycle of uranium or plutonium (the once-through fuel cycle). The following alternative fuel cycle options have also been assessed insofar as they relate to spent fuel transport.

Uranium Recycle Only

For this fuel cycle option spent fuel would be reprocessed rather than treated as a waste.

Recycle of Uranium and Plutonium

Spent fuel is also reprocessed for this fuel cycle option and would not be treated as a waste.

6.2.2 Truck Transport of Spent Fuel

Almost half the nuclear reactors currently operating, under construction, or licensed for construction in the U.S. do not have rail spurs at the reactor site. Spent fuel shipments from many of these reactors are expected to be made by intermodal casks such as the IF-300. Other reactors are expected to use truck transport for shipment of irradiated fuel to interim storage. For planning purposes, in the year 2000 it is assumed that 10% of spent fuel shipped from reactors to interim storage facilities will be transported by truck. Truck casks will not be used for shipping canistered fuel from interim storage to final isolation.

Information about currently licensed truck casks for spent fuel is given in Table 6.2.1. Shipping information is given in Table 6.2.2.

6.2.2.1 Truck Casks for Shipment of Spent Fuel

NFS-4 (NAC-1)

The NFS-4 cask of Nuclear Fuel Services is a water-filled truck cask designed to transport one PWR or two BWR assemblies.[23,24] Figure 6.2.5 shows it mounted on a special lightweight trailer. Approximate loaded cask weight is 23 MT (50,000 lb). The NAC-1 cask of Nuclear Assurance Corporation is similar to the NFS-4.[25] Two NFS-4 casks and four NAC-1 casks have been built and used for truck transport of spent fuel assemblies.

The cask has an overall length of 5.44 m (214 in.) and a diameter of 0.96 m (38 in.). The cask cavity has a length of 4.52 m (178 in.) and a diameter of 0.34 m (13.5 in.). Interchangeable fuel baskets provide the cask with a capacity of one PWR or two BWR fuel assemblies.

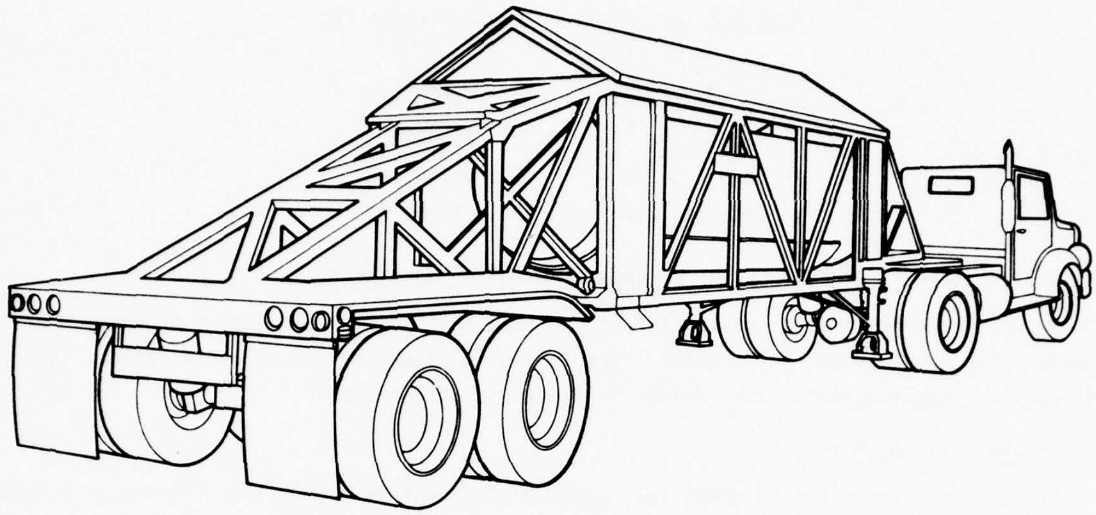

FIGURE 6.2.5. NFS-4 Cask and Special Trailer

The primary cask cavity consists of a nominal 0.8-cm (5/16 in.) stainless steel pressure shell surrounded by a lead gamma shield 17 cm (6-5/8 in.) thick and a stainless steel penetration barrier 3.2 cm (1-1/4 in.) thick. Neutron shielding is provided by a borated water-antifreeze solution contained in a compartmentalized tank 11.4 cm (4-1/2 in.) thick which surrounds the cask. An expansion chamber for the shield tank accommodates temperature changes of the water-antifreeze solution. Impact limiters of stainless steel-sheathed balsa wood at each end of the cask give protection from impact damage.

The container has a single lid, attached with high-strength bolts and sealed with Teflon O-rings. The closure requires a lifting spider, special tools and O-ring pressure test equipment. Two valve-type drain closures are provided. Reported times required for loading and unloading are about 10 hr each.

Heat rejection is by convection through the water coolant in the cavity to the inner wall, conduction to the neutron shield, convection to the outer wall, and convection plus radiation to the atmosphere. Maximum heat rejection capacity is 11.5 kW. Maximum design conditions for the inner cavity during normal transport [i.e., 54°C (130°F) direct sunlight, still air, maximum fuel burnup, minimum fuel cooling period] are 174°C (345°F) and 150 psig. Normal pressure upon receipt is almost always less than 5 psig, however, so the design is quite conservative. The primary cavity is designed to withstand temperature and pressure conditions of 278°C (532°F) and 984 psig under the fire accident condition [1/2 hr at a temperature of 802°C (1475°F)].

NLI 1/2

National Lead Industries' NLI 1/2 cask[26] is a helium-filled truck cask designed to transport one PWR or two BWR fuel assemblies. Approximate loaded cask weight is 22 MT (48,000 lb). Two of these casks have been built in the United States.

The cask has an overall length of 4.93 m (194 in.) and a diameter of 1.08 m (42-1/2 in.), including cooling fins. The cask cavity has a length of 4.52 m (178 in.) and a diameter of 34 cm (13-3/8 in.). The cask can be used with an optional wall liner 0.6 cm (1/4 in.) thick which provides an additional level of containment if desired. Use of the optional liner reduces the cavity diameter to 32 cm (12-5/8 in.). Interchangeable fuel baskets provide a capability of transporting one PWR or two BWR fuel assemblies.

The primary cask cavity consists of a nominal 1.3 cm (1/2 in.) stainless steel pressure shell surrounded by a gamma shield composed of 7 cm (2-2/3 in.) of depleted uranium and 5.4 cm (2-1/8 in.) of lead and a 2.5 cm (1 in.)-thick stainless steel penetration barrier. Neutron shielding is provided by 12.7 cm (5 in.) of water. The water jacket surrounding the cask also carries cooling fins which are welded to the outside of the jacket. An expansion tank is provided for the water jacket. Attached to each end of the cask is a conical structure which serves as impact limiter.

The cask closure head consists of a ring forging whose center section is filled with depleted uranium sandwiched between stainless steel plates. The head is bolted to the cask body and sealed with two elastomer O-ring gaskets.

Heat rejection is by convection through the helium coolant to the inner cavity wall, conduction to the neutron shield, convection to the outer wall, and convection plus radiation from the finned surface. Maximum design heat rejection capacity is 10.6 kW. Maximum fuel temperature during normal transport is conservatively estimated at 545°C (1013°F). Normal maximum design pressure is 120 psig when the inner container is used and 22.5 psig when it is absent. Maximum fuel temperature during the fire accident condition is 594°C (1102°F). The cask has a pressure rating of 543 psig at 454°C (850°F) when the inner container is used, and 264 psig at 454°C (850°F) when it is absent.

TN-8

The TN-8 cask of Transnuclear, Inc. is a truck cask designed to transport three PWR assemblies in an air atmosphere.[27,28] Approximate loaded cask weight is 36 MT (80,000 lb). The TN-8 will normally travel by truck under overweight restrictions, although two casks could be placed together on a single rail car. The cask was licensed by the Atomic Energy Commission (AEC) in 1974. No casks have yet been built in the United States although casks of the same design are presently used in Europe.

The cask has an overall length of 5.54 m (18-1/6 ft) with impact limiters attached, and an outside diameter of 1.71 m (67.5 in.). The inner cavity is 4.27 m (14 ft) long and consists of three separate compartments 23 x 23 cm (9.1 x 9.1 in.) for the individual fuel elements.

The main gamma shield is a layer of lead 18.5 cm (7.3 in.) thick located between the stainless steel-lined inner cavity and the carbon steel outer shell. Together with other materials, the total equivalent lead thickness is 23 cm (9 in.). It reduces the maximum gamma dose rate to 35 mR/hr at the cask surface. The neutron shield is 15 cm (6 in.) of borated solid resin. Balsa wood located inside removable covers and fixed end drums provides impact protection.

Heat rejection is via conduction through the cask body to the outer wall, with convection and radiation from copper cooling fins on the outside. Maximum heat rejection capacity is 35.5 kW. During normal operation the maximum temperature of the inner shell(s) is 115°C (239°F). Pressure is stated to be atmospheric during transport. The cavity design pressure is 110 psig.

TN-9

The TN-9 cask of Transnuclear, Inc. is a truck cask designed to transport seven BWR assemblies in an air atmosphere.[27,28] The TN-9 is similar to the TN-8 except that it has seven compartments, each 4.52 m (14-5/6 ft) long and 15 x 15 cm (5.9 x 5.9 in.) in cross section. For shipment of BWR assemblies, heat removal capacity is 24.5 kW.

6.2.2.2 Secondary Wastes from Truck Shipment of Spent Fuel

Secondary wastes are generated in cask loading and unloading operations at nuclear reactors and interim storage facilities. Generation and treatment of these wastes are discussed in the receiving and shipping facility descriptions in other sections of this document.

6.2.2.3 Emissions from Truck Shipment of Spent Fuel

Information about radiation fields and heat generation rates for both rail and truck casks for shipment of spent fuel is given in Table 6.2.2 and in Section 6.2.1.3.

6.2.2.4 Decommissioning of Spent Fuel Truck Casks

The useful life of a spent fuel cask is estimated to be 20 to 30 years. Decommissioning is assumed to be accomplished by appropriate decontamination procedures followed by disposal as non-TRU waste.

6.2.2.5 Postulated Accidents for Truck Shipment of Spent Fuel

Casks for truck shipment of spent fuel are designed and constructed to Type B package specifications. Truck casks, like spent fuel rail casks, are expected to withstand all but very severe, highly unusual accidents (see Section 6.2.1.5).

Several transportation accidents have been reported[29,30] in which Type B truck casks have been subjected to severe mechanical environments or to fire. None of these accidents has resulted in a release of package contents or in an excessive external radiation level. The following are examples[30] of severe accident environments to which Type B truck casks have been subjected:

While attempting to negotiate a wide turn the driver of a vehicle carrying a spent fuel cask swerved to avoid colliding with an approaching vehicle, lost control, and overturned on a roadway. As a result, the cask assembly was thrown into a ditch, traveling more than 100 ft before coming to rest. The driver of the vehicle was fatally injured as a result of impact and crush forces. Minor damage to the outer thermal insulation was suffered by the cask, but there was no release of cask contents and no increase in radiation level. The cask was recovered and returned to service after repairs.

A truck carrying five cylindrical shipping containers filled with uranium scrap suffered loss of both rear wheels while traveling at about 50 mph. The truck crashed through a guard rail and continued down the steep slope on the side of the roadway for several hundred yards, eventually coming to rest upside down in a drainage ditch. The five shipping containers were torn loose from their tiedowns within the vehicle; however, there was no release of package contents.

Tables 6.2.11, 6.2.12, and 6.2.13 give postulated minor, moderate and severe accident scenarios for truck shipment of spent fuel from reactors to interim spent fuel storage facilities. Accident scenarios for truck shipment of spent fuel are similar to those for rail shipment of spent fuel (see Section 6.2.1.5) except for scenarios involving loss of mechanical cooling. Because truck casks carry only about one-tenth the payload of rail casks, truck casks do not employ auxiliary cooling systems. Truck casks are not expected to reach rod perforation temperatures except under very unusual extended fire conditions.

Expected frequencies for accidents postulated in Tables 6.2.11, 6.2.12, and 6.2.13 are based on the accident probabilities per vehicle mile from Section 6.1.3 and on total shipment miles calculated from data in Table 6.2.2.

For purposes of environmental consequence analysis, the material releases associated with accident number 6.2.13 in Table 6.2.12 has been selected as an umbrella source term. (The concept of an umbrella source term is explained in Section 3.7.) This means that the releases from this accident is the largest in its source term category. The environmental consequences of these accidents are described in DOE/ET/0029. Accidents are cross indexed with their appropriate umbrella source term in Appendix A, Section 3.

6.2.2.6 Cost of Truck Transport of Spent Fuel

Estimates, in mid-1976 dollars, of capital, freight, and total unit costs for truck transport of spent fuel from reactors to an independent storage basin and packaging facility are presented in this section.

Capital Cost of Spent Fuel Truck Cask System

The capital cost of the waste transportation system for truck shipment of spent fuel is estimated to be $900,000 with an accuracy range of ±25%. In making this estimate the system is treated as purchased equipment supplied repetitively by qualified vendors on a competitive basis. The estimate reflects manufacturer's profits and development, engineering, sales, overhead, and other similar expenses in addition to material and manufacturing costs.

The capital cost estimate includes costs for the complete transportation system including the cost of the cask, trailer, tiedown system, and sun shield. Items excluded from the estimate include truck tractors and the owner's technical, procurement, and general administrative expenses associated with the purchase of the transport system.

The accuracy range reflects the uncertainties in the optimum design of a spent fuel truck cask. Additionally, the range reflects the spread in pricing normally experienced in quotes from manufacturers of specialized heavy equipment.

TABLE 6.2.11. Spent Fuel Truck Cask Minor Accidents

Accident No. and Description	Sequence of Events	Safety System	Release
6.2.9 – Truck collision or overturn accident involves spent fuel cask. Expected frequency ~2 per year	1. Collision or overturn accident occurs. 2. Truck leaves roadway and may overturn. 3. Confinement barriers of cask remain intact. 4. Accident is reported to local and Federal officials. 5. Cask recovered.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site and recover cask. 3. Confinement barriers of cask contain all radioactive materials.	None
6.2.10 – Truck collision or overturn accident and 1/2 hour (or less) fire involves spent fuel cask. Expected frequency ~0.2 per year.	1. Collision or overturn accident occurs. 2. Truck leaves roadway and may overturn. 3. Spent fuel cask is involved in 1/2 hour (or less) fire. 4. Confinement barriers of cask remain intact. 5. Accident is reported to local and Federal officials. 6. Cask recovered.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site and recover cask. 3. Confinement barriers of cask contain all radioactive materials.	None
6.2.11 – Undetected leakage of coolant from cask cavity (and/or surface contamination washoff). Expected frequency ~2 per year.	1. A defective gasket or valve results in a leak of not more than 0.001 cc/sec (below detection threshold). 2. Leakage of cavity coolant occurs for the duration of the trip. 3. Cavity coolant contains some radioactivity due to transportation with a small percentage of fuel rods with failed cladding.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Checks made on cask prior to release for shipment would detect larger leaks.	For casks which use water as a cavity coolant, assume 0.1 Ci of gross fission product activity in coolant (1 μCi/cc). In 1 day an undetected leak of 0.001 cc/sec would release about 90 μCi of activity. Estimate 1% of the released activity in the liquid is dispersed in the form of an aerosol.

TABLE 6.2.12. Spent Fuel Truck Cask Moderate Accidents

Accident No. and Description	Sequence of Events	Safety System	Release
6.2.12 — Truck collision or overturn results in loss of neutron shielding from spent fuel truck cask. Expected frequency $\sim 2 \times 10^{-2}$ per year.	1. Collision or overturn accident occurs. 2. Truck leaves roadway and overturns. 3. Cask leaves trailer. 4. Water jacket is ruptured and neutron shield water is lost. 5. Accident is reported to local and Federal officials. 6. Cask recovered.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site, minimize personnel exposure, and recover cask. 3. Remaining confinement barriers of cask remain intact and contain all radioactive materials.	No radioactive material release. Neutron emission rates are: 4.36×10^7 n/sec-g $^{242}CmO_2$ 1.10×10^7 n/sec-g $^{244}CmO_2$ Mean neutron energy is 4 MeV.
6.2.13 — Fire accompanying accident causes truck cask cavity to overpressurize; relief valve operates. Expected frequency $\sim 2 \times 10^{-2}$ per year.	1. Collision or overturn accident occurs at moderate speed. 2. Spent fuel cask is involved in 1/2 hour (or longer) fire. 3. Cask cavity overpressurizes and pressure relief valve operates to relieve pressure. 4. 0.1% of cavity coolant is released from cask due to operation of pressure relief valve. 5. Cavity coolant contains some radioactivity due to transportation with a small percentage of fuel rods with failed cladding. 6. Accident is reported to local and Federal officials. 7. Area decontaminated. 8. Cask recovered.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site, minimize personnel exposure, and recover cask. 3. Confinement barriers of cask remain intact and limit release to a small percentage of cavity coolant.	Venting of cask occurs at ground level without off-gas system control. Assuming 0.25% of fuel rods exhibit failed cladding, the following release fractions apply: Isotope — Release Fraction ^{85}Kr — 7.5×10^{-7} ^{129}I — 2.5×10^{-7} ^{3}H — 2.5×10^{-7} For casks which use water as cavity coolant, assume an additional 0.1 Ci of gross fission product activity in the coolant. Venting of a "wet" cask results in release of 1×10^{-4} Ci of gross fission products. Release fractions apply to cask inventory of 0.4 MTHM. Use release period of 1 hour. Tables 3.3.8 and 3.3.10 describe the activity spectrum of 0.5-year-old fuel.

TABLE 6.2.13. Spent Fuel Truck Cask Severe Accidents

Accident No. and Description	Sequence of Events	Safety System	Release
6.2.14 – Collision or overturn causes spent fuel truck cask to be subjected to severe impact and fire. Expected frequency ~2 x 10^{-5} per year.	1. Collision or overturn accident occurs at high speed. 2. Spent fuel cask strikes massive object and decelerates almost *instantaneously*. 3. Cask is involved in a fire which lasts *longer than 1 hour*. 4. Closure head seal breached by gasket failure or by cask lid bolts being sheared off. 5. 10 of fuel rods are perforated and 100 of cavity coolant is released. 6. Accident is reported to local and Federal officials. 7. Cask sealed. 8. Area decontaminated. 9. Cask recovered.	1. Interagency radiological assistance personnel available to assist local public safety and carrier officials to control site, minimize personnel exposure, *and recover cask*. 2. Only a small opening exists in cask. Accident does not result in gross breach of cask containment.	Ground-level release occurs with following release fractions: *Isotope* — *Release Fraction* ^{85}Kr — 3×10^{-2} ^{129}I — 1×10^{-2} ^{3}H — 1×10^{-2} Mixed fission products — 1×10^{-7} Actinides — 1×10^{-7} Release fractions apply to cask inventory of 0.4 MTHM. Use release period of 2 hours. Tables 3.3.8 and 3.3.10 describe the activity spectrum 0.5-year-old fuel.

Cask Use Charge for Spent Fuel Truck Cask

A cask use charge was calculated for a spent fuel truck cask using the above capital cost and an annual operating cost for maintenance of 2% of the capital cost. Private ownership and a use factor of 80% or 292 days per year were assumed. Section 3.8 gives the details for calculating the levelized charges. On this basis the daily use charge was calculated to be $740 per day. Table 6.2.14 summarizes the cask-day calculations. Although daily use charges per ton of fuel are higher for truck shipments compared to rail, the shorter travel time for truck transport compensates for that disadvantage and the total use charges are about the same for either truck or rail transport.

TABLE 6.2.14. Calculation of Cask-Days per Trip

Waste Shipment	One-Way Distance, miles	Casks Carried/ Trip	Round-Trip Travel Time/Cask, days	Round-Trip Turnaround Time/Cask, days	Cask-days/ Trip
Spent fuel truck	1000	1	3	2	5

Unit Cost Estimate for Truck Transport of Spent Fuel

Table 6.2.15 shows the derivation of the haulage charge calculations for all truck shipments. Table 6.2.16 shows the unit cost estimate for truck transport of spent fuel. The unit cost is the sum of the unit cask use charge and the unit haulage charges.

TABLE 6.2.15. Haulage Charges for Truck Shipment of Spent Fuel[31]

Waste Type	Round-Trip Distance, miles	Basic Fee, $/mile	Special Equipment,[a] $/mile	Second Driver, $/mile	Holding for Load, Unload,[b] $/mile	Total	
						$/mile	$/trip
Spent fuel to IFSF	2000	0.95	0.19	0.15	0.19	1.48	3000

a. Based on deadhead charges for moving special equipment (tractor) 400 miles at $0.75/mile.
b. Assuming 10-hr detention charges at $15/hr at each end.

TABLE 6.2.16. Unit Cost Estimate for Truck Shipment of Spent Fuel to Storage Basin

Source	One-Way Distance, miles	Cask Capacity, MTHM	Casks Carried/ Trip	Daily Cask Use Charges, $/MTHM/day	Cask-days/ Trip	Total Cask Use Charge, $/kg HM	Haulage Charge, $/kg HM	Total Unit Cost, $/kg HM
PWR	1000	0.461	1	1630	5	8.15	6.50	14.65 ±25%
BWR	1000	0.377	1	2000	5	10.00	8.00	18.00 ±25%

Although on a unit cost basis truck shipments appear to cost about the same as rail shipments, these figures do not take into account the large increase in cask handling capacity that would be required at origins and destinations if all shipments were made by truck. This would require a substantial additional investment in handling facilities. Thus for the reference spent fuel transportation system, it has been assumed that 45% of the fuel is shipped in NLI 10/24 rail casks, 45% of the fuel is shipped in IF-300 rail casks, and 10% of the fuel is shipped in truck casks. As shown in Table 6.2.2, the number of truck cask shipments is about half of the total number of shipments under these assumptions.

6.2.7.7 Construction Requirements for Spent Fuel Casks

It is estimated that about 3,000 kg of steel and 20,000 kg of lead will be used in fabricating a spent fuel truck cask. These approximate quantities of basic construction materials do not include allowances for canisters, trailers, support systems, or other appurtenances.

6.2.2.8 Effects of Fuel Cycles on Truck Shipment of Spent Fuel

The reference process (once-through cycle) for truck shipment of spent fuel assumes no recycle of uranium or plutonium. The following alternative fuel cycle options have also been assessed insofar as they relate to spent fuel transport.

Uranium Recycle Only

For this fuel cycle option spent fuel would be reprocessed rather than treated as a waste.

Recycle of Uranium and Plutonium

Spent fuel is reprocessed for this fuel cycle option also and would not be treated as a waste.

6.2.3 Barge Transport of Spent Fuel

Movement of irradiated nuclear fuel from offshore (floating) nuclear power plants will probably require transport by barge. The spent fuel may be transported by barge directly to a shore facility for treatment or storage. Alternatively, barge transport would be to the nearest land-based transfer point; here the material would be reloaded for overland shipment by truck or rail.

Casks used for barge shipment of spent fuel would probably be similar to those used for rail transport. A barge might carry as many as four casks per shipment. An environmental statement related to the proposed construction of floating nuclear power plants[32] contains an estimate of the annual number of barge shipments of spent fuel required from a representative 1100-MWe offshore power station. Table 6.2.17 shows these data.

Barge transportation of nuclear fuel is an alternative to truck or rail shipment where both the reactor and the processing or storage facility are on navigable waterways. Movement of nuclear materials by barge suggests high payloads and low tariffs. However, gains in

TABLE 6.2.17. Annual Shipments of Spent Fuel from Offshore Nuclear Power Station, One 1100 MWE PWR(33)

Operation	Approximate No. of Shipments per Year Barge	Land
Spent fuel		
1. Offshore power plant to shore transfer point	2 to 5 barges	
2. Shore transfer point to treatment or storage		60 trucks or 10 rail cars

payload and decreased tariffs could be offset by the costs of longer transit times. Estimates indicate as much as a tenfold increase in transit times for barge shipments as compared to truck or rail shipments. Since cask use charges are a major component of shipping costs, the additional time is a strong economic deterrent.

6.2.3.1 Safety Aspects of Barge Transport

Barge shipments of spent fuel and radioactive waste from offshore nuclear power stations must meet the same standards and regulations for packaging and carriage of radioactive material as are in effect for land-based shipments, namely, the regulations of the Nuclear Regulatory Commission (10 CFR 71) and the Department of Transportation (49 CFR 170-189). Also the Coast Guard requires that shipments of irradiated fuel be made in Type A barges, i.e., double-walled vessels that are "less likely to sink." Casks transported by barge must be tied down so that, if the barge does sink, the casks will stay with the barge for ease of location and recovery. Insofar as possible, barge shipments must be made in water no more than 150 m deep, again for ease of recovery in case of sinking.

Development of equipment for water transport of spent fuel has been reported.[33] The emphasis of this research has been to develop a water transport vehicle safer and more reliable than the conventional barge-tow boat arrangement. The research has also resulted in design of a vessel that can be beached in the shallow water encountered at reactor sites. This eliminates the need for costly dock or landing facilities.

6.2.3.2 Postulated Accidents for Barge Shipment of Spent Fuel

From accident statistics provided by the U.S. Coast Guard it has been estimated[32] that the accident rate for barges is about 9.7 accidents per million barge miles. This is about four times greater than the truck accident rate and about six times greater than the rail accident rate (measured in accidents per car mile).

Few data are available on the severity of barge accidents. Barges travel only a few miles per hour; therefore, the deceleration experienced in accidents would be small. However, because of the large mass of the vehicle and cargo, severe impact forces could be encountered. Cargo could also be subjected to crush forces of large magnitude in accidents. Fires at sea can last from a few minutes to a few days. Perhaps the most extreme case would be when a cargo

vessel collides with a bulk liquid carrier transporting millions of gallons of highly flammable fuel and both vessels stay together and burn on the surface. A recent study[32] estimates that less than 2% of barge accidents result in severe mechanical damage and that fire occurs in less than 10% of these severe accidents.

In barge shipments of spent fuel, cargo damage resulting from mechanical forces and fire would be comparable to that experienced by rail shipments under similar accident environments. Accident scenarios are given in Section 6.2.1.5 for rail shipments of spent fuel.

Radionuclides released to the ocean tend to be dispersed rapidly by the natural mixing of ocean waters. However, biological mechanisms exist in the marine food chain for concentrating some radioactive substances. The primary potential hazard to man following a release of radioactive material into the ocean is probably through the eating of radioactively contaminated seafood.

Except under very unusual circumstances in which the cask could not be immediately located or was submerged in extreme depths, a cask which was lost at sea could probably be recovered with normal salvage equipment. For unusual circumstances the technology currently exists that allows the location and recovery of large objects from almost any point on the ocean floor. This technology includes the use of scanning sonar systems, manipulating arms, and of television and color cameras which can be towed slowly over the ocean floor. Miniature submarines known as Deep Submergence Vehicles have been developed that have greatly enhanced the capability for recovery of objects from the ocean floor.

A cask accidentally dropped into water during barge transport is unlikely to be affected adversely unless the water is deep. The closure seal on a typical spent fuel cask includes a gasket designed to withstand an internal working pressure of about 400 psi. Static seals are normally not as pressure-resistant in the direction opposite to the design differential pressure. It may therefore be assumed that internal seals will begin to leak at about 300 psi external pressure. A hydrostatic pressure of 300 psi would occur at a water depth of about 220 m (720 ft). This implies that for a cask lost on the continental shelf, leakage past the seals might occur. However, hydrostatic pressures encountered on the continental shelf are too low to cause the cask wall to collapse. Once the cask closure is breached, the fuel rods are exposed to the hydrostatic pressure. However, in an operating reactor fuel rods are exposed to internal pressures of about 2000 psi. Thus, fuel rods are unlikely to suffer collapse due to hydrostatic pressure on the continental shelf.

If cask closure is breached, the Zircaloy fuel clad would be exposed to sea water. Zirconium and Zircaloys are exceptionally stable in seawater, exhibiting a corrosion resistance similar to that of titanium.[34] Corrosion rates can be estimated from data for titanium alloys exposed to seawater as determined in U.S. Naval Engineering Laboratory tests.[35] These tests resulted in no visible corrosion on specimens of alloys, which indicates a rate less than 0.001 mil per year. Certain isolated areas of unusually rapid corrosion, such as welds, were observed to corrode at rates of about 0.01 mil per year. Thus it is probable that the rate at which corrosion would breach fuel rods exposed to seawater would be measured on a time scale of a few thousand years.

If a spent fuel cask fell in water for a distance of 200-300 ft, it could acquire sufficient velocity to bury itself in the ocean sediments upon impact. Because thermal conductivity of marine sediments is low, a cask remaining buried several days would experience a rise in temperature (from radioactive decay heat). This could result in perforation of the fuel pins and possible cask failure. However, the marine sediments in which the cask is buried would provide an additional barrier to release of radioactive fission products to the ocean.

Should a cask accidentally be dropped during loading or unloading from a barge, the main effect would probably be limited to that of rather severe damage to the barge. It is possible that a cask could penetrate through the barge. In this event it would fall into relatively shallow water from which it could be recovered with normal salvage operations. The most significant effect of such an accident would be some economic loss.

6.2.4 Physical Protection and Safeguard Requirements for Transport of Spent Fuel

Because spent fuel might be the target of theft or sabotage during shipment, physical protection and safeguard requirements should be considered. First, several characteristics of the fuel itself make it an undesirable target for theft. Among these are its intense radioactivity, size and weight (1/4 to 1/2 ton per assembly), and relatively low plutonium content per assembly. Also, to extract the plutonium from fuel assemblies in even the simplest possible manner would require a rather complex facility that provides for remote operations of a cutup saw, dissolver, transfer pumps or lifts, several stages of chemical purification, off-gas filtration (to avoid early detection of a clandestine operation), and liquid waste disposal capability. In addition, a considerable amount of technical knowledge and craftsmanship must be available to the group that would construct and operate such a facility. Even with prior plutonium processing experience, the system shakedown with the first fuel assembly would probably require several weeks at least.

Since spent fuel is more accessible when being transported than when at a fixed site and since successful penetration of the fuel canisters would contaminate the immediate vicinity of the incident, the fuel must be protected from assault.[a] The casks themselves offer a good deal of protection. Cask covers, encumbered with cooling equipment and weighing about a ton, require a crane for removal. This process would consume considerable time. Once the cover is off, the fuel assemblies must be removed. Then, should the saboteur intend to explode the fuel, the tight packing of the elements and the level of radioactivity would hamper introduction of explosives into the cask.[36] Finally, the localized direct radiation is potentially lethal to those who might try to remove the fuel elements from the cask by other techniques.

A penetration of the cask with firearms or a shaped charge might be possible. If the element cladding were broken, the volatile fission products and a small amount of the nonvolatile fission products would be released. If liquid coolant were being used, it might be

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a. 10 CFR Part 73, Section 73.6b, notes that spent fuel in transit is exempted from requirements outlined in Sections 73.30 through 73.36, "Physical Protection of Special Nuclear Material in Transit." This exemption is based upon the substantial protection provided by the spent fuel shipping cask against removal or dispersal of its radioactive contents.

lost, and the fuel temperature would gradually increase over several hours. This, too, might cause element cladding ruptures and, again, release of volatile fission products. The consequences of such actions are analagous to those described in accident descriptions 6.2.6, 6.2.7, 6.2.8 and 6.2.14.

The fuel must also be protected from possible hijacking of a truck and cask together. Once thieves have driven the truck away from the authorized routing, there might be ample time to remove the fuel from the cask and disperse it or, perhaps, to attempt extortion. A deterrent to dispersal would be the difficulty of working with the intensely radioactive material once it is minus the shielding provided by the cask. An extortion attempt would necessarily be limited to the length of time law enforcement personnel would need to locate the stolen property. Such material would be detectable by aerial radiation surveys and the fact that detection would be imminent would deter any lengthy extortion scheme.

The probability of a successful theft and its hazards to the public health and safety are minimal. Thirty years of successful spent fuel transporting backs this up. After an extensive analysis of potential accidents as well as of actual cases, the risk to the public from accidents is judged to be very small. Sabotage and theft potentialities do not appear to increase that risk. However, if a judgment is made in the future that the safeguards requirements need to be increased, the measures specified in 10 CFR 73 for transport of strategic special nuclear material can be implemented.

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6.3 TRANSPORTATION OF HIGH-LEVEL WASTE

6.3 TRANSPORTATION OF HIGH-LEVEL WASTE

High-level waste is the primary waste generated when spent fuel is reprocessed. It contains nearly all the fission products, about 0.5% of the uranium and plutonium, and most of the other transuranics originally present in the irradiated fuel.

High-level waste is currently not being commercially solidified. Casks designed specifically for shipment of solidified high-level waste have not been built. Shipping containers for high-level waste are expected to resemble the casks currently available for truck and rail shipment of spent fuel. Prime considerations in the shipment of both high-level waste and spent fuel are the dissipation of radioactive decay heat and protection from high radioactive dose rates. High-level waste shipping casks will satisfy the criteria for Type B packaging.

To permit dissipation of decay heat, it would be advantageous to postpone shipment of the solidified waste till near the end of the 10-year storage time currently allowed by Federal regulations. This is clearly evident from Figure 5.1.1, which shows the heat generation rate of high-level waste as a function of age.

6.3.1 Rail Transport of High-Level Waste

High-level waste could be shipped by truck or rail. Because of their greater payload capacity it is assumed that rail casks will be used for solidified waste shipments from fuel reprocessing plants to final isolation facilities. Conceptual rail cask designs have been published by Oak Ridge National Laboratory[1,2] and by Pacific Northwest Laboratory (PNL).[3] Both designs provide for transport of multiple waste canisters in a single cask and incorporate many features of contemporary spent fuel cask designs. The PNL design has been used as the basis for shipping calculations in this report.

Table 6.3.1 presents an example calculation of the number of rail shipments of solidified high-level waste from fuel reprocessing plants to permanent disposal at a Federal repository. Information in the table is based on shipping 6.5-year-old waste and on the projected requirements for the uranium plus plutonium recycle option in the year 2000 (see Section 2.1).

6.3.1.1 Rail Cask for Shipment of High-Level Waste

Solidified high-level waste is assumed to be contained in disposable stainless steel canisters with welded closures. A conceptual canister would be 3.05 m (10 ft) long with a diameter such that the heat generation rate of the contained waste would not exceed a specified limit that depends on the geologic medium of the repository where the waste is to be emplaced. These limits are 3.2 kW per canister for salt, 1.7 kW for granite, 1.2 kW for shale and 1.3 kW.

A conceptual high-level waste rail cask design published by Battelle Pacific Northwest Laboratory[3] has been used as the basis for this study. The cask is a lead-filled double-walled stainless steel cylinder weighing about 100 MT (220,000 lb). Nine 0.30-m (12-in.) diameter high-level waste canisters can be accommodated in an aluminum insert that fits into the cask cavity. By changing the aluminum insert, the cask could also be made to accommodate thirteen 0.25-m (10-in.) diameter, or twenty 0.20-m (8-in.) diameter, or thirty-six 0.15 m (6-in.) diameter waste canisters. Each of these configurations would transport the same quantity of

SHLW. Thus, regardless of the canister heat generation limit, the number of required SHLW shipments does not change. The total number of waste canisters per cask shipment would always be limited by the cask maximum thermal design load of 50 kW.

The cask is transported on an exclusive use, six-axle rail car. The gross shipping weight of the loaded cask and rail car is about 150 MT (330,000 lb).

TABLE 6.3.1. Shipping Information for Rail Shipment of Solidified High-Level Waste from Fuel Reprocessing Plants to Final Isolation in the Year 2000[a]

Number of canisters shipped	2610
Number of canisters per cask	13
Number of casks shipped	200
Radioactivity, per cask[b]	8×10^6 Ci
Thermal power per cask [b]	32 kW
Assumed shipping distance, one way	1500 miles
Transport time, one way	10 days

a. Calculations are based on shipping vitrified waste in 0.25-m (10-in.) diameter canisters.
b. Assume shipment at age 6.5 years.

Table 6.3.2 gives the basic structural details of the conceptual high-level waste cask. The cask is shown in Figure 6.3.1. Leaf springs position the waste canisters in holes in the cavity insert, and energy-absorbing plugs are attached to the ends of the canisters. In addition to the gamma shielding provided by the lead and steel structural material, neutron shielding is furnished by a water jacket which surrounds the cask body. Impact protection is provided by circumferential fins surrounding the cask body and by radial fins on the ends of the cask.

Two covers (inner and outer) are provided for cask closure. The covers are of stainless steel with depleted uranium for gamma protection. They are secured to the cask body by high-strength studs.

The cask will dissipate up to 50 kW of internally generated heat. Heat dissipation is aided by the fins surrounding the cask body. Auxiliary cooling of the cask is not required.

6.3.1.2 Secondary Wastes from Rail Shipment of High-Level Waste

Secondary wastes are generated in cask loading and unloading operations at fuel reprocessing plants and at Federal facilities for interim storage or final isolation. Generation and treatment of these wastes are discussed in the descriptions of those facilities.

6.3.1.3 Emissions from Rail Shipment of High-Level Wastes

Shipments of solidified waste would be subject to radiation dose rate limits prescribed by the U.S. Department of Transportation (see Section 6.1.1). For shipments in closed vehicles, current Federal regulations impose dose rate restrictions of 200 mrem/hr at the external surface of the vehicle, 10 mrem/hr at any point 1.8 m (6 ft) from the vehicle, and 2 mrem/hr at any normally occupied position in the vehicle.

TABLE 6.3.2. Details of the Conceptual Rail Cask for Transport of Solidified High-Level Waste

Length of structural shell, excluding fins	4 m (13.2 ft)
Diameter of structural shell, excluding fins	2.1 m (6.8 ft)
Length of cask cavity	3.3 m (10.8 ft)
Diameter of cask cavity	1.5 m (5.0 ft)
Cask weight	100 MT (220,000 lb)
Maximum thermal design load	50 kW
Cavity shell	
Material	Stainless steel
Thickness	1.9 cm (3/4 in.)
Gamma shield	
Material	Lead
Thickness	20 cm (8 in.)
Outer shell	
Material	Stainless steel
Thickness	5 cm (2 in.)
Neutron shield	
Material	Borated Water
Thickness	10 cm (4-in.)
Impact absorbers	
Internal	Honeycomb
External	Steel fins

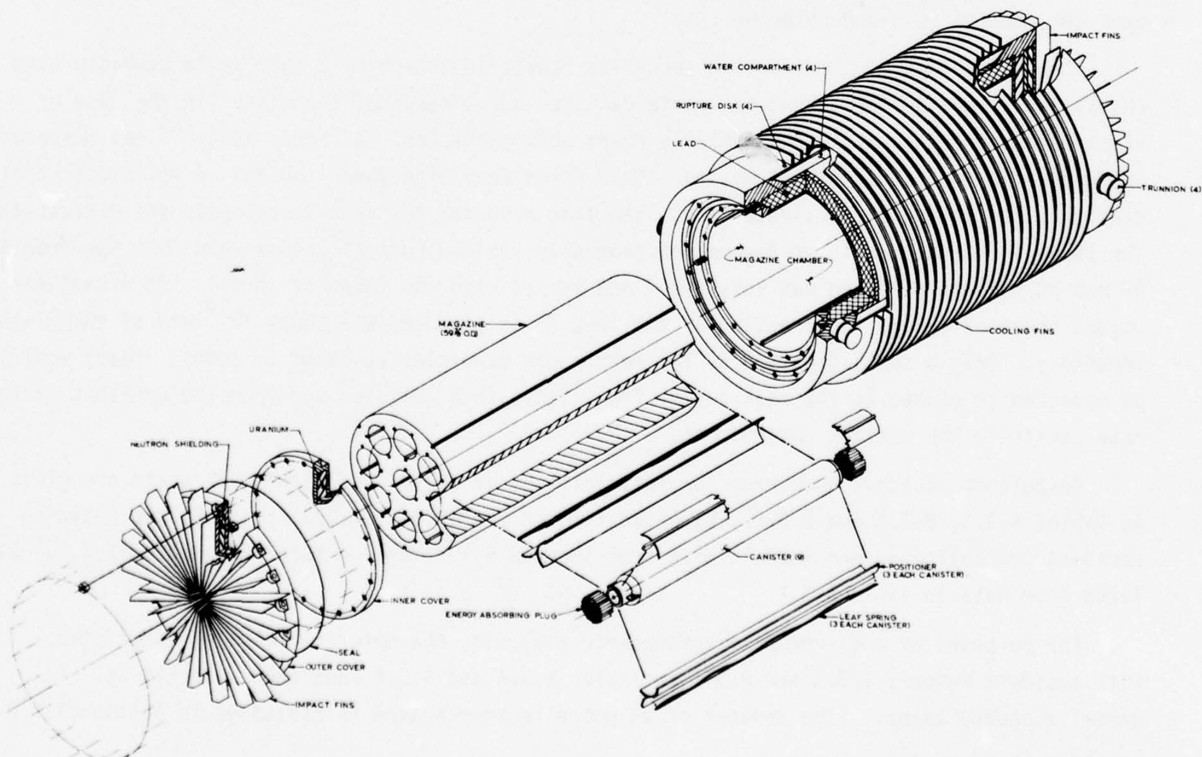

FIGURE 6.3.1. Conceptual Cask for Rail Shipment of Solidified High-Level Waste

The rate of heat release to the air from the conceptual high-level waste cask would be about 40 kW. This heat release rate is less than the rate at which waste heat is released from a 100-hp truck engine operating at full power, i.e., about 50 kW. Federal regulations limit the temperature of the accessible surface of a shipping container to 82°C (180°F). Hence the temperature of the air contacting a cask shipment would be increased a few degrees, but the temperature of the air a few feet from the shipment would be unaffected.

6.3.1.4 Decommissioning of High-Level Waste Rail Casks

The useful life of a high-level waste cask is estimated to be 20 to 30 years. Decommissioning is assumed to be accomplished by appropriate decontamination procedures followed by disposal as non-TRU waste.

6.3.1.5 Postulated Accidents for Rail Shipment of High-Level Waste

Rail casks for shipment of solidified high-level waste are expected to resemble casks currently available for rail shipment of spent fuel. Hence, operating experience with spent fuel casks should be applicable to high-level waste transportation. Transport experience with spent fuel casks is described in Sections 6.1.2 and 6.2.1.5.

Federal regulations(4) require shipment of high-level waste as a solid. Several methods for solidifying the waste are being investigated,(5) including calcination and vitrification. In a severe transportation accident, the solid form of the waste would be a significant barrier to any release of radioactivity. In addition, the waste would be doubly contained by the canister and the shielded shipping cask.

A mechanical environment severe enough to result in breach of a cask would probably also result in failure of the high-level waste canister and breakup of the glass (in the case of vitrified waste) into small, potentially respirable particles. A recent study(6) has measured the production of respirable (i.e., sub-10-μ) fines from high-speed impacts of stainless steel canisters containing borosilicate glass like that proposed for high-level waste solidification. The study found that canister damage was generally limited to small cracks and that the fraction of sub-10-μ fines produced was very small and varied with the speed of impact. At a canister impact speed of 60 mph the fraction of sub-10-μ fines produced was about 10^{-2} wt% of the glass inventory. Only a small fraction of the respirable particles produced by severe impact would be expected to escape to the cask cavity from cracks in a canister and from the confines of the cask cavity to the outside environment.

Postulated accident scenarios for rail shipment of solidified high-level waste are given in Tables 6.3.3, 6.3.4 and 6.3.5. Expected frequencies of postulated accidents are based on accident probabilities per vehicle mile from Section 6.1.3, and on total shipment miles calculated from data in Table 6.3.1.

For purposes of environmental consequence analysis, the material releases associated with accident numbers 6.3.4 and 6.3.5 in Tables 6.3.4 and 6.3.5 have been selected as umbrella source terms. (The concept of an umbrella source term is explained in Section 3.7.)

TABLE 6.3.3. Solidified High-Level-Waste Rail Cask Minor Accidents

Accident No. and Description	Sequence of Events	Safety System	Release
6.3.1 — Train derailment involves high-level-waste cask. Expected frequency ∿0.3/yr.	1. Derailment accident occurs. 2. Railcar leaves track and may overturn. 3. Confinement barriers of cask remain intact. 4. Accident is reported to local and Federal officials. 5. Railroad crews restore car to track.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site and recovery cask. 3. Confinement barriers of cask contain all radioactive materials.	None
6.3.2 — Train derailment and 1/2 hr (or less) fire involves high-level-waste cask. Expected frequency ∿0.03/yr.	1. Derailment accident occurs. 2. Railcar leaves track and may overturn. 3. Cask is involved in 1/2 hr (or less) fire. 4. Confinement barriers of cask remain intact. 5. Accident is reported to local and Federal officials. 6. Railroad crews restore car to track.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site and recover cask. 3. Confinement carriers of cask contain all radioactive materials.	None
6.3.3 — Unusual transport condition erodes cask surface. Expected frequency ∿1/yr.	1. Cask is shipped with surface contamination which is within limits prescribed by Federal regulations. 2. Shipment encounters heavy rain storm or violent wind and dust storm. 3. Exceptional weather erodes cask surface. 4. 10% of residual surface contamination is removed by weathering.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Casks are decontaminated and monitored prior to shipment to assure that removable surface contamination levels are within limits prescribed by Federal regulations.	4×10^{-6} Ci of mixed fission products and actinides. Tables 3.3.9 and 3.3.14 describe the activity spectrum

TABLE 6.3.4. Solidified High-Level-Waste Rail Cask Moderate Accidents

Accident No. and Description	Sequence of Events	Safety System	Release
6.3.4 — Collision or derailment results in loss of neutron shielding from high-level-waste cask. Expected frequency $\sim 3 \times 10^{-3}$/yr.	1. Collision and/or derailment accident occurs at moderate speed. 2. Railcar overturns. 3. Cask leaves railcar. 4. Water jacket is ruptured and neutron shield water is lost. 5. Accident is reported to local and Federal officials. 6. Cask recovered.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site, minimize personnel exposure, and recover cask. 3. Remaining confinement barriers of cask remain intact and contain all radioactive materials.	No radioactive material release. Neutron emission rates are: 4.36×10^{7} n/sec-g $^{242}CmO_2$ 1.10×10^{7} n/sec-g $^{244}CmO_2$ Mean neutron energy is 2 MeV.

TABLE 6.3.5. Solidified High-Level-Waste Rail Cask Severe Accidents

Accident No. and Description	Sequence of Events	Safety System	Release
6.3.5 — Collision or derailment subjects high-level-waste cask to severe impact and fire. Expected frequency $\sim 3 \times 10^{-6}$/yr.	1. Collision and/or derailment accident occurs at high speed. 2. High-level-waste cask strikes massive object and decelerates almost instantaneously. 3. Cask is involved in a fire lasting more than 1 hr. 4. Closure head seal breached by gasket failure or by cask lid bolts being sheared off. 5. Mechanical shock causes small cracks in half of high-level-waste canisters. 6. Breakup of waste glass results in production of small percentage of respirable fines. 7. Accident is reported to local and Federal Officials. 8. Cask sealed. 9. Area decontaminated. 10. Cask recovered.	1. Interagency radiological assistance personnel available to assist local public safety and carrier officials to control site, minimize personnel exposure, and recover cask. 2. Only a small opening exists in cask. Accident does not result in gross breach of containment.	For glass waste a mixed fission product and actinide release in the form of sub-10 μ fines is postulated with a release fraction of 5×10^{-8}. For calcine a mixed fission product and actinide release is postulated with a release fraction of 5×10^{-6}. Cask contains 6.5-year-old waste from processing 27.4 MTHM of spent fuel. Assume release occurs at ground level and lasts 0.25 hr. Tables 3.3.9 and 3.3.14 describe activity spectrum of the waste inventory.

This means that the releases from these accidents are the largest in their respective source term categories. The environmental consequences of these accidents are described in DOE/ET/0029. Accidents are cross indexed with their appropriate umbrella source terms in Appendix A, Section 3.

Postulated accident scenarios do not include the effects of fire because of its low probability and the low probability that the radioactivity release would be affected. Data[7] on train derailment and collision accidents show that about 1% of these accidents are accompanied by fire and that only about 10% of these fires last 2 hr or longer. The waste forms being considered (calcine and borosilicate glass) have both been heated to at least 900°C (1650°F) during processing. It is unlikely that this temperature would be exceeded in any credible accident involving a fire. Thermal decomposition or a release of volatile material will not be a problem if the material is later heated to a lower temperature. Also pressure buildup in these waste forms appears insignificant up to 900°C except for the volatilization of water accidentally added to a filled, baked out canister prior to closure welding.

6.3.1.6 Cost of Rail Shipment of High-Level Waste

Estimates have been made, in mid-1976 dollars, of capital, freight, and total unit costs for rail transport of solidified high-level waste.

Capital Cost of High-Level Waste

The capital cost of the waste transportation system for rail transport of solidified high-level waste is estimated to be $2.5 million with an accuracy range of ±40%. In making this estimate the system is treated as purchased equipment supplied repetitively by qualified vendors on a competitive basis. The estimate reflects manufacturer's profits and development, engineering, sales, overhead, and other similar expenses in addition to material and manufacturing costs.

The capital cost estimate includes costs for the complete transportation system including the cost of the cask, rail car, tiedown system, heat exchangers, and sun shield. Items excluded from the estimate include the owner's technical, procurement, and general administrative expenses associated with the purchase of the transport system.

The accuracy range reflects the uncertainties in the optimum design of a high-level waste rail cask. Additionally, the range reflects the spread in pricing normally experienced in quotes from manufacturers of specialized heavy equipment.

Cask Use Charge for High-Level Waste Cask

A cask use charge was calculated for a solidified high-level waste (SHLW) cask using the above capital cost and an annual maintenance cost of 2% of the initial capital cost. Private ownership and a use factor of 80% or 292 days per year were assumed. Section 3.8 gives the method for calculating levelized charges. On this basis the daily cask use charge is calculated to be $2200/day. Table 6.3.6 gives the cask-day calculations for SHLW shipments.

TABLE 6.3.6. Calculation of Cask-Days per Trip for SHLW

Waste Shipment	Distance, miles	Casks Carried/Trip	Round-Trip Travel Time/Cask, days	Round-Trip Turnaround Time/Cask, days	Cask-Days/Trip
SHLW - rail	1500	1	20	4	24

Unit Cost Estimate for Rail Transport of SHLW

Table 6.3.7 shows the unit cost estimate for rail transport of SHLW. The unit cost is the sum of the unit cask use charge and the unit freight charge. The freight charge calculations are based on Figure 6.2.3. The estimated unit cost for special train requirements is shown in the last column of Table 6.3.7. This calculation makes the same assumptions as used for spent fuel special train shipments.

TABLE 6.3.7. Unit Cost Estimate for Rail Shipment of SHLW to Federal Repository

HLW Solidification Process	Distance, miles	Cask Capacity, Equiv. MTHM	Canisters[a]	Number of Casks/Trip	Daily Cask Use Charge, $/MTHM/day	Cask-Days/Trip	Total Cask-Use Charge, $/kg HM	Freight Charge, $/kg HM	Total Unit Cost, $/kg HM	Unit Cost Assuming Special Train Requirement $/kg HM
Calcination	1500	26.35	9	1	84	24	2.00	0.90	2.90 ±50%	4.30
Vitrification (reference)	1500	27.40	9	1	81	24	1.90	0.90	[illegible].80 ±50%	4.20

a. .3 m dia x 3 m (1 ft dia x 10 ft) canisters.

6.3.1.7 Construction Requirements for High-Level Waste Rail Casks

An estimate has been made of the approximate quantities of basic construction materials employed in fabricating a high-level waste rail cask. These are 25,000 kg of steel and 75,000 kg of lead. They do not include allowances for canisters, rail cars, support systems, or other appurtenances.

6.3.1.8 Effects of Fuel Cycle Options on Rail Shipment of Solidified High-Level Waste

The reference process for rail shipment of solidified high-level waste assumes reprocessing of spent LWR fuel and recycling the retrieved uranium and plutonium. The following alternative fuel cycle options have also been assessed insofar as they relate to transport of solidified high-level waste.

No Recycle

Eliminating the reprocessing operation does away with the generation of high-level wastes. Accordingly, no high-level waste transportation is required.

Uranium Recycle Only, with Plutonium to a Repository

This alternative is expected to generate about the same amount of solidified high-level waste as in the uranium and plutonium recycle case. Transportation requirements are expected to be the same as for the reference process.

Uranium Recycle Only, with Plutonium to High-Level Waste (HLW)

Incorporation of plutonium in the high-level waste may necessitate modifications to the HLW rail cask. Cask modifications which might be required to accommodate the shipment of plutonium with the waste could include:

1. a thicker water shield because of the increased neutron dose rate
2. fewer canisters per cask because of the increased heat load
3. modifications to the cask to assure criticality control during shipment of the waste.

A criticality safety analysis has been conducted for the transportation of canisters of solidified high-level waste assuming the incorporation of plutonium in the waste. The analysis indicates no insurmountable criticality safety problems with the transportation system, although there are still some unresolved questions.

Two alternative forms of solidified waste were considered: fluidized-bed calcine product in 0.2 m (8 in.) diameter canisters, and waste glass in 0.3 m (12 in.) diameter canisters. Although k_∞ = 1.313* for the calcine, the values k_{eff}* for full canisters 3.05 m (10 ft) long are only 0.207 and 0.145 for full concrete and water reflection, respectively. Thus the handling of single calcine canisters should be critically safe. Some restrictions on the stacking of large numbers of bare calcine canisters will be required. However, these would not be severe, since 50 of them stacked together would be subcritical. To establish the criticality safety of a shipping cask loaded with calcine canisters, calculations were done for infinitely long cask models having a shield of 20 cm (8 in.) of lead (or depleted uranium) and 12.5 cm (5 in.) of water. For conservative simplicity the aluminum canister spacer was removed, and the entire 1.5 m (5 ft) diameter cavity was assumed to be filled with calcine containing discarded plutonium. This conservative configuration is subcritical, having k_{eff} = 0.945 and 0.894 for lead and depleted uranium, respectively.

Criticality safety control for the waste glass canisters depends on whether or not the plutonium oxide tends to concentrate due to precipitation and settling during the glass-making operation. With the plutonium homogeneously distributed, k_∞ is much less than 1, and no additional criticality control is required. However, if the distribution of plutonium is too nonhomogeneous, it may be necessary to reduce the mass of plutonium in each canister or the number of canisters in the shipping cask. As discussed in detail in Section 4.1.1.9, this criticality question cannot be fully resolved until experimental information on the solubility of plutonium in waste glass becomes available.

6.3.2 Truck Transport of High-Level Waste

Truck transport of solidified high-level waste is assumed to be feasible on the same basis as truck transport of spent fuel. Truck casks for high-level waste shipment would probably be limited to one canister per cask and would resemble casks currently available for

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* The number of second generation fissions per fission of a nucleus by a first generation neutron is called the multiplication factor and is denoted by k. The value of k for a system that is infinitely large (a system from which there is no leakage of neutrons) is called k_∞. For an assembly of finite size, k is usually called k_{eff}, the effective multiplication factor.

truck shipment of spent fuel. Since the payload (measured in terms of kilograms of waste transported per metric ton of cask weight) is relatively low for truck casks, and since rail facilities would be available at both the reprocessing plant and at interim storage and final isolation facilities, the reference transport mode for high-level waste shipments is assumed to be rail.

6.3.3 Physical Protection and Safeguard Requirements for Transport of Solidified High-Level Waste (HLW)

High-level waste contains nearly all of the fission products, about 0.5% of the uranium and plutonium, and most of the other TRU originally present in the irradiated fuel. (High-level waste in the case of uranium-only recycle differs from the above and is discussed separately in this section.) This waste would not be a credible source of strategic quantities of plutonium because the plutonium concentration is very low and it is not practical to extract it from the waste. However, the waste could be considered a target for sabotage or theft because it is a source of highly radioactive material that may be dispersed as an act of terrorism. The attractiveness of radioactive materials for these forms of terrorism is considered low, as discussed in Section 3.9. In addition, HLW would be quite inaccessible for either theft or rupture of a container in an act of sabotage because of the ruggedness of the containers (see 6.3.1.1 and 6.3.2) and the fact that the material would be in a solidified form such as a glass or sintered ceramic material (see 4.1).

Protection during transport would be enhanced by the fact that the railroad shipping casks weigh about 100 MT and are designed to withstand severe accidents. Consequently, the casks would also be quite resistant to small arms fire and explosives. In addition, the consequences of penetration of a cask would be a very minor release of radioactive material at the site of the sabotage act (see Tables 6.3.4 and 6.3.5).

Removal of the waste from a cask outside of a special shielded facility would not be credible. Cask covers could not be removed by hand because of their weight. Should the saboteur intend to explode the waste, the close packing of the waste canisters and their level of radioactivity would hamper introduction of explosives into the cask. In addition, the localized direct radiation is potentially lethal to those who might try to remove the canisters from the casks by other techniques.

Should the HLW be transported by truck, hijacking of a truck and cask together might be possible. Once thieves had driven the truck away from the authorized routing, there might be ample time to remove waste from the cask and disperse the contents or, perhaps, to attempt extortion. A deterrent to dispersal would be the difficulty of working with the intensely radioactive material when it is removed from the shielding provided by the cask. An extortion attempt would necessarily be limited to the length of time law enforcement personnel would need to locate the stolen property. Such material would be detectable by aerial radiation surveys. The fact that detection would be imminent at any time would be a deterrent to any lengthy extortion scheme.

Safeguarding of high-level waste shipments would not be required by present regulations because of the high radiation levels of the material (see 10 CFR Part 73, Section 63.6(b)). The sabotage and theft potentialities do not appear to require it. However, if a judgment is made in the future that the safeguards requirements need to be increased, the measures specified in 10 CFR 73 for transport of strategic special nuclear material (Sections 73.30 through 73.36) can be implemented.

High-level waste from the uranium-only recycle process would contain plutonium at moderate concentrations. Although the radiation levels of the solidified waste would be very high, recovery of the plutonium in pure form in a shielded processing facility is feasible. The attractiveness and accessibility of this waste for theft or sabotage would be very similar to that of spent fuel. The safeguarding considerations discussed above in 6.2.4, "Physical Protection and Safeguards Requirements for Transport of Spent Fuel", are applicable to this case.

REFERENCES FOR SECTION 6.3

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2. J. J. Perona and J. O. Blomeke, A Parametric Study of Shipping Casks for Solid Radioactive Wastes. ORNL-TM-3651, Oak Ridge National Laboratory, Oak Ridge, TN, February 1972.

3. P. L. Peterson and R. E. Rhoads, Conceptual Design of a Shipping Container for Transporting High-Level Waste by Railroad. PNL-2244 Battelle, Pacific Northwest Laboratories, Richland, WA, November 1977.

4. Title 10, Code of Federal Regulations, Part 50, Appendix F (10 CFR 50, Appendix F).

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6.4 TRANSPORTATION OF FUEL RESIDUE

6.4 TRANSPORTATION OF FUEL RESIDUE

Fuel residues are solid residues from fuel reprocessing operations in which fuel bundles have been sheared in short lengths and the fuel pellets acid leached from the cladding. The residues include short lengths of fuel cladding with some residual fuel, massive end fittings, fuel support grids, and assorted springs, spacer elements, and fuel bundle support rods. The radioactivity of cladding wastes arises both from neutron-induced isotopes and from the fission products and actinides present in the small amount of fuel that remains with the cladding. In fuel cycle options which involve the reprocessing of spent fuel, the cladding hulls and assembly hardware would be sealed in metal canisters and shipped in massive casks to Federal repositories for interim storage or final isolation.

Cladding wastes generated to date in commercial fuel reprocessing operations have been disposed of by onsite burial. Casks designed specifically for rail or truck shipment of fuel bundle residues have not been built. These casks are expected to resemble casks currently available for shipment of spent fuel but to be somewhat simpler in design because heat removal requirements would be reduced and neutron shielding would not be required. It is assumed that transport systems for both rail and truck shipment of cladding wastes can be designed which meet all applicable Federal regulations for transport of radioactive materials.

6.4.1 Rail Transport of Fuel Residues

Fuel residues could be shipped by rail or truck. Because rail casks have greater payload capacity, it is assumed that they will be used for shipments from fuel reprocessing plants to interim storage or final isolation.

Table 6.4.1 presents an example calculation of the number of rail shipments of cladding hulls and assembly hardware from fuel reprocessing plants to a Federal repository. This information is based on the conceptual canister and cask described in Section 6.4.1.1 and on projected requirements for spent fuel reprocessing in the year 2000 (see Section 2.1). Shipping requirements are given only for the no-compaction treatment option.

Treatment of fuel residues to sort hulls and hardware and to reduce the volume of hulls by mechanical compaction or melting can reduce the number of fuel residues shipments by about a factor of 2 relative to the number required for the no compaction option.

6.4.1.1 Rail Cask for Fuel Residues

The conceptual waste canister for fuel residues shipments is assumed to have a diameter of 76 cm (30 in.) and a length of 3.05 m (10 ft). It would be made of stainless steel with a wall thickness of about 0.6 cm (1/4 in.), end thicknesses of 1.3 cm (1/2 in.), and welded closures.

Casks for the transport of cladding wastes would conform to the requirements for Type B packaging and would provide adequate gamma shielding to meet the radiation dose requirements of 49 CFR 173.393. Neutron shielding would not be required. Decay heat loads are low enough that cooling fins would probably also not be required. Impact protection for the cask could be provided by steel-clad, balsa impact limiters.

TABLE 6.4.1. Rail Shipment of Fuel Residues from Fuel Reprocessing Plants in Year 2000[a]

Canisters shipped annually	1790
Canisters per cask	3
Casks shipped annually	597
Radioactivity per cask, Ci	1×10^5
Thermal power per cask, W	795
Assumed shipment distance, one way, miles	1500
Transport time, one way, days	10

a. Calculations are based on the no compaction treatment option.

For planning purposes a rail cask has been postulated that would transport three canisters. The conceptual cask is assumed to be a lead-filled, double-walled stainless steel cylinder weighing about 65 MT (143,000 lb). An insert would serve to position the three canisters inside the cask cavity and would act as a heat conduction path from the waste canisters to the inner surface of the cavity wall. The cask would not be pressurized. Table 6.4.2 provides basic cask design parameters.

6.4.1.2 Secondary Wastes from Rail Shipment of Fuel Residues

Secondary wastes may be generated in cask loading and unloading operations at fuel reprocessing plants and at Federal facilities for interim storage or final isolation. These wastes are included in the wastes treated at those facilities.

6.4.1.3 Facility Effluents from Rail Shipment of Fuel Residues

Shipments of fuel residues would be subject to radiation dose rate limits prescribed by the U.S. Department of Transportation (see Section 6.1.1). For shipments in closed vehicles, current Federal regulations impose dose rate restrictions of 200 mrem/hr at the external surface of the vehicle, 10 mrem/hr at any point 1.8 m (6 ft) from the vehicle, and 2 mrem/hr at any normally occupied position in the vehicle.

The total rate of heat release from a cask shipment of cladding wastes would be a few kilowatts. Such a heat release would not affect the temperature of the air a few feet from the shipment.

6.4.1.4 Decommissioning of Rail Casks for Shipment of Fuel Residues

The useful life of a hulls cask is estimated to be 20 to 30 years. Decommissioning is assumed to be accomplished by appropriate decontamination procedures followed by disposal as non-TRU waste.

6.4.1.5 Postulated Accidents for Rail Shipment of Fuel Residues

Rail casks for shipment of fuel residues are expected to resemble spent fuel casks and will be built to Type B package standards. Hence, operating experience with spent fuel casks

TABLE 6.4.2. Details of the Conceptual Rail Cask for Transport of Fuel Residues

Length of structural shell	3.9 m (152 in.)
Diameter of structural shell	2.2 m (88 in.)
Length of cask cavity	3.4 m (132 in.)
Diameter of cask cavity	1.8 m (70 in.)
Number of canisters transported	3
Cask weight	
Net weight	65 MT (143,000 lb)
Loaded weight[a]	74.5 MT (164,000 lb)
Cavity shell	
Material	Stainless steel
Thickness	2 cm (3/4 in.)
Gamma shield	
Material	Lead
Thickness	15 cm (6 in.)
Outer shell	
Material	Stainless steel
Thickness	5 cm (2 in.)
Impact absorbers	
Internal	Honeycomb
External	Steel-clad balsa wood

a. Assumes canisters loaded with 1340 kg of uncompacted hulls and hardware plus 1340 kg of sand to fill void spaces.

should be applicable to fuel residue transport. Transport experience with spent fuel casks is described in Sections 6.1.2 and 6.2.1.5.

Postulated accident scenarios for rail shipment of fuel residues are given in Tables 6.4.3 and 6.4.4. Expected frequencies of postulated accidents are based on accident probabilities per vehicle mile from Section 6.1.3 and on total shipment miles calculated from data in Table 6.4.1.

6.4.1.6 Cost of Rail Shipment of Fuel Residues

Estimates have been made, in mid-1976 dollars, of capital, freight, and total unit costs for rail shipments of fuel residues.

Capital Cost of Cask System for Fuel Residues

The capital cost of the waste transportation system for rail transport of cladding hulls and assembly hardware is estimated to be $600,000 with an accuracy range of ±40%. In making this estimate, the system is treated as purchased equipment supplied repetitively by qualified vendors on a competitive basis. The estimate reflects manufacturers' profits and development, engineering, sales, overhead, and other similar expenses in addition to material and manufacturing costs.

TABLE 6.4.3. Fuel Residue Rail Cask Minor Accidents

Accident No. and Description	Sequence of Events	Safety System	Release
6.4.1 — Derailment involves fuel residues cask. Expected frequency ~1/yr.	1. Derailment occurs. 2. Railcar leaves track and may overturn. 3. Confinement barriers of cask remain intact. 4. Accident is reported to local and Federal officials. 5. Railroad crews restore car to track.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to aid local public safety and transport carrier officials to control site and recover cask. 3. Confinement barriers of cask contain all radioactive materials.	None
6.4.2 — Derailment and 1/2 hr (or less) fire involves fuel residues cask. Expected frequency ~0.1/yr.	1. Derailment occurs. 2. Railcar leaves track and may overturn. 3. Fuel residues cask is involved in 1/2 hr (or less) fire. 4. Confinement barriers of cask remain intact. 5. Accident is reported to local and Federal officials. 6. Railroad crews restore car to track.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to aid local public safety and transport carrier officials to control site and recover cask. 3. Confinement barriers of cask contain all radioactive materials.	None
6.4.3 — Unusual transport condition erodes cask surface. Expected frequency ~2 per year.	1. Cask is shipped with surface contamination which is within limits prescribed by Federal regulations. 2. Shipment encounters heavy rainstorm or violent wind and dust storm. 3. Exceptional weather erodes cask surface. 4. Weathering releases 10% of residual surface contamination.	1. Radiation warning signs on cask caution onlookers to keep distance. 2. Casks are decontaminated and monitored prior to shipment to assure that removable surface contamination levels are within limits prescribed by Federal regulations.	4×10^{-6} Ci of mixed fission products and actinides. Tables 3.3.9, and 3.3.15 describe the activity spectrum.

TABLE 6.4.4. Fuel Residue Rail Cask Severe Accidents

Accident No. and Description	Sequence of Events	Safety System	Release
6.4.4 — Collision or derailment subjects fuel residues cask to severe impact and fire. Expected frequency $\sim 1 \times 10^{-5}$/yr.	1. Collision and/or derailment at high speed. 2. Fuel residues cask strikes massive object and decelerates almost instantaneously. 3. Cask is involved in fire more than 1 hr. 4. Closure head seal breached by gasket failure or by cask lid bolts being sheared off. 5. Small perforations occur in one waste canister. 6. Accident is reported to local and Federal officials. 7. Cask sealed. 8. Area decontaminated. 9. Cask recovered.	1. Interagency radiological assistance personnel available to aid local public safety and transport carrier officials to control site and recover cask. 2. Only a small opening exists in cask. Accident does not result in gross breach of cask containment.	Respirable particle release occurs with a release fraction of 1×10^{-6}. Cask contains fuel residue from about 4 MTHM of fuel. See Table 3.3.28 to define the activity spectrum. Assume release occurs at ground level and lasts 2 hr.

The capital cost estimate includes costs for the complete transportation system, including the cost of the cask, rail car, tiedown system, and sun shield. Items excluded from the estimate include railway locomotives and the owner's technical, procurement, and general administrative expenses associated with the purchase of the transport system.

The accuracy range reflects the uncertainties in the optimum design of a rail cask for shipment of cladding wastes. It also reflects the spread in pricing normally experienced in manufacturers' quotes for specialized heavy equipment.

Cask Use Charge

A cask use charge was calculated for the residues cask using the above capital cost and an annual maintenance cost of 2% of the initial capital cost. Private ownership and a use factor of 80% or 292 days per year were assumed. Section 3.8 gives the method for calculating levelized charges. On this basis, the daily cask use charge is $530 per day. Table 6.4.5 gives the cask-day calculations for fuel residue shipments.

Unit Cost Estimate for Rail Transport of Fuel Residues

Table 6.4.6 shows the unit cost estimates for rail transport of three alternative fuel residue forms. The unit cost is the sum of the unit cask use charge and the unit freight charge. The freight charge calculations are based on Figure 6.2.3 data. The estimated unit factor of 80% or 292 days per year were assumed. Section 3.8 gives the method for calculating levelized charges. On this basis, the daily cask use charge is $530 per day. Table 6.4.5 gives the cask-day calculations for fuel residue shipments.

The effect of waste volume reduction in the treatment processes on the transportation costs can be clearly seen when the units are expressed in terms of kgHM of spent fuel processed.

TABLE 6.4.5. Calculation of Cask-Days per Trip for Fuel Residues

Waste Shipment	Distance, miles	Casks Carried/ Trip	Round-Trip Travel Time/ Cask, days	Round-Trip Turnaround Time/Cask, days	Cask-Days/ Trip
Fuel residues - rail	1500	1	20	4	24

TABLE 6.4.6. Unit Cost Estimate for Rail Shipment of Fuel Residues

Process	Distance, miles	Capacity, Equiv. MTHM	Canisters	Casks/ Trip	Daily Cask Use Charge, $/MTHM/day	Cask-Days/ Trip	Total Cask-Use Charge, $/kg HM	Freight Charge, $/kg HM	Total Unit Cost, $/kg HM	Unit Cost Assuming Special Train Requirement, $/kg HM
Untreated (reference)	1500	12.5	3	1	42	24	1.00	2.00	3.00	5.80
Mechanical compaction	1500	21.3	3	1	25	24	0.60	1.10	1.70	3.30
Melting	1500	31.1	3	1	17	24	0.40	0.80	1.20	2.30

6.4.1.7 Construction Requirements for Rail Casks for Shipment of Fuel Residues

It is estimated that about 16,000 kg of steel and 49,000 kg of lead would be the basic construction materials employed in fabricating a rail cask for shipment of fuel residues. These quantities do not include allowances for canisters, rail cars, support systems, or other appurtenances.

6.4.1.8 Effects of Fuel Cycle Options on Rail Shipment of Fuel Residues

The reference process for rail shipment of fuel residues assumes reprocessing of spent LWR fuel and recycling the retrieved uranium and plutonium. The following alternative fuel cycle options have also been assessed insofar as they relate to transport of fuel residues.

No Recycle

Eliminating the fuel reprocessing operation does away with the generation of fuel residues. Accordingly, no fuel residues transportation is required.

Uranium Recycle Only

This alternative is expected to generate about the same amount of fuel residues as in the uranium and plutonium recycle case. Transportation requirements are expected to be the same as for the reference process.

6.4.2 Truck Transport of Fuel Residues

Truck casks for shipment of cladding wastes would probably resemble casks currently available for truck shipment of spent fuel and would be limited to one canister per cask. The casks would be constructed to Type B package standards and would provide adequate gamma shielding to meet the radiation dose requirements of 49 CFR 173.393.

The conceptual truck cask is assumed to be a lead-filled, double-walled stainless steel cylinder weighing about 19.5 MT (43,000 lb). Highway weight limitations would probably necessitate use of a special lightweight trailer for shipment of a loaded cask. Special high-strength, low-weight semi-trailers are currently used[1] for shipment of spent fuel truck casks.

The payload (measured in terms of kilograms of waste transported per metric ton of cask weight) is significantly lower for truck casks than it is for rail casks. Reprocessing plants and Federal repositories may prefer rail shipments to truck shipments since less cask handling is required. Therefore the reference mode for shipment of fuel residues is assumed to be by rail.

6.4.3 Physical Protection and Safeguard Requirements for Transport of Fuel Residue

Fuel residue waste would not be an attractive theft or sabotage target because its metallic form would render it difficult to disperse and its radioactive materials content would be

relatively low (Section 3.3.4). The consequences of the transport accidents shown in Tables 6.4.3 and 6.4.4 would be representative of the potential consequences of an act of sabotage. No additional physical protection and safeguards measures seem to be necessary beyond that afforded by the packaging and containment required and provided for safety protection (10 CFR 71).

REFERENCES FOR SECTION 6.4

1. W. C. Benson, "The Development of High Strength, Low Weight Semi-Trailers for Transportation of Nuclear Fuel." Proceedings of the Fourth International Symposium on Packaging and Transportation of Radioactive Materials, CONF 740901, Miami Beach, FL, September 1974.

6.5 TRANSPORTATION OF PLUTONIUM

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6.5 TRANSPORTATION OF PLUTONIUM

Plutonium is separated from spent fuel at a reprocessing plant. In the fuel cycle option that involves recycle of uranium only, the plutonium would be treated as a waste and shipped to a Federal interim storage facility. Federal regulations to take effect in June 1978[1] require that plutonium be shipped as a solid (assumed to be plutonium oxide). All material shipped must be packaged in a separate inner container placed within outer packaging that meets the standards set forth in 10 CFR 71. The separate inner container must not release plutonium when the entire package is subjected to the normal and the accident test conditions in Appendices A and B of 10 CFR 71.

Shipments of plutonium in excess of 2 kg must be made in accordance with Nuclear Regulatory Commission approved transport plans which provide for the physical protection of special nuclear material in transit.[2] Federal regulations for the transport of plutonium require the use of secure transportation systems with the following features:

- NRC-approved preplanned carrier procedures
- hand-to-hand receipts
- tamper-indicating seals on containers and cars
- no cargo transfers enroute
- continuous visual surveillance of the shipment
- periodic radio communication between transport vehicle and carrier dispatcher
- armed escorts or specially designed transport vehicles with disabling features.

6.5.1 *Truck Transport of Plutonium*

Current shipping practice is to transport plutonium by sole-use truck, and it is anticipated that this will continue to be the primary shipping mode. Table 6.5.1 presents an example calculation of the number of truck shipments of plutonium oxide from fuel reprocessing plants to a Federal interim storage facility. The information is based on requirements for the uranium-recycle-only fuel cycle option projected for the year 2000. The plutonium shipping container used as basis for the shipment information is described in Section 6.5.1.1.

TABLE 6.5.1. Truck Shipment of Plutonium Oxide from Fuel Reprocessing Plants to a Federal Interim Storage Facility, Year 2000

Annual production, PuO_2	6.3×10^4 kg
PuO_2/package	32 kg
Packages/yr	2190
Packages/vehicle	10
Radioactivity/package	1.3×10^5 Ci
Thermal power/package	560 W
Vehicle shipments/yr	219
Assumed distance, one way	1500 miles
Time, one way	2 days

6.5.1.1 Plutonium Shipping Cask

Plutonium is in Transport Group I, which requires Type B packaging for any shipment containing more than 1 mCi of material. Because plutonium is a fissile material, all containers for shipping plutonium must conform to the packaging requirements for fissile material. Shipping containers are presently available for those forms of plutonium that require little or no shielding and have a relatively low heat generation rate.[3] A recently developed conceptual container[4] (the AGNS PPP-1 shipping container) would incorporate both gamma and neutron shielding and has a greater heat dissipation capacity than is available in present packaging. For planning purposes, this section describes this container as the reference package for shipments of plutonium.

The AGNS PPP-1 shipping container is designed to ship 32 kg of plutonium oxide powder. Eight kilograms of PuO_2 would be held in each of four thin-walled stainless steel canisters. The canisters would be placed in a primary pressure vessel that would be contained within a Type B overpack. Figure 6.5.1 shows the primary pressure vessel, and Figure 6.5.2 depicts the total plutonium package. Table 6.5.2 lists the package specifications. The thin-walled stainless steel canisters are 15 cm (6 in.) in diameter and 31 cm (12 in.) high. The canisters are

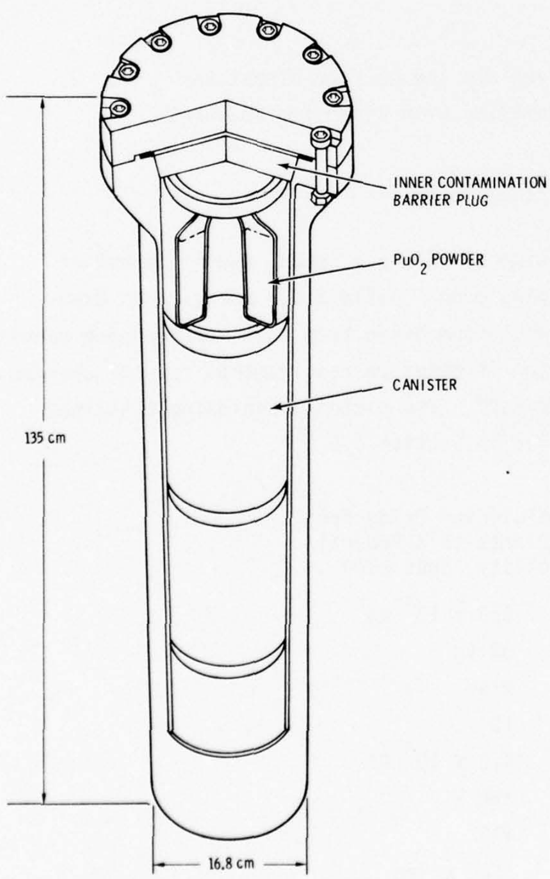

FIGURE 6.5.1. Plutonium Oxide Container Primary Pressure Vessel

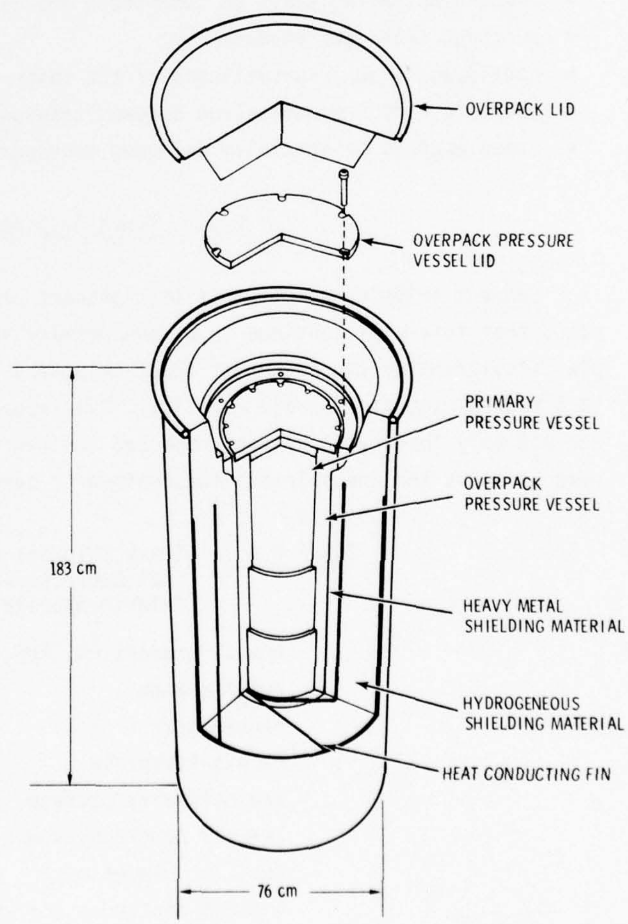

FIGURE 6.5.2. Plutonium Oxide Shipping Container

TABLE 6.5.2. Package Specifications for the AGNS PPP-1 Plutonium Oxide Shipping Container

Package type	Pressure vessel inside Type B overpack
Overpack height	1.83 m (72 in.)
Overpack diameter	0.76 m (30 in.)
Structural material	Stainless steel
Gamma shield	Lead
Neutron shield	Solid organic
Pressure vessel height	1.35 m (53 in.)
Pressure vessel diameter	16 cm (6.62 in.)
Net weight of PuO_2	32 kg
Total weight of package	1636 kg (3600 lb)

designed with an annular powder volume to limit powder centerline temperatures. A vent on each canister is designed to allow gases such as water vapor or radiolytic decomposition products of water to escape from the canister while providing particulate filtration to contain the oxide powder.

The primary pressure vessel would contain four canisters. The vessel is 16 cm (6.62 in.) in diameter and 1.35 m (53 in.) high and is constructed of 300 series stainless steel. The primary pressure vessel lid uses an elastomeric O-ring seal and is secured by twelve 7/8-in. bolts. A vent is provided to permit controlled release of gas pressure either prior to shipping or after receipt. However, the pressure vessel is designed to withstand pressures from the combined effects of accident condition temperatures and total radiolytic decomposition of moisture which may be adsorbed on the oxide powder. Design pressure of the primary pressure vessel is 230 psig.

The Type B overpack is fabricated from 300 series stainless steel and has outer dimensions of 0.76 m (30 in.) diameter by 1.83 m (72 in.) high. The overpack provides radiation shielding and fire and impact protection. Gamma shielding is provided by a lead annulus. Neutron shielding is provided by 20 cm (8 in.) of a solid organic material sandwiched between the inner and outer walls of the overpack. The inner liner of the overpack, with lid attached, serves as a secondary pressure vessel with a design pressure of 160 psig. The inner liner lid is sealed with an elastomeric O-ring seal and secured by sixteen 1-in. bolts. As shown in Figure 6.5.2, an additional overpack cover is employed, secured by sixteen 5/8-in. bolts.

Impact protection from both drop and puncture is provided by the neutron shield material in the Type B overpack. The sides of the overpack and the cover are fitted with temperature-activated vent devices. These would release any gases which might be formed in the neutron shield during a thermal excursion such as an accidental fire.

Total weight of the plutonium oxide shipping container (including contents) is about 1636 kg (3600 lb). The calculated heat dissipation capability of the cask under licensing conditions (260°C maximum canister centerline temperature) is 621 W. The package is designed to be handled by a fork lift truck. The maximum number of packages assumed to be transported by sole-use truck in a single shipment is 10.

At the Federal interim storage facility, the PuO_2 powder would be stored inside the primary pressure vessel. The Type B overpack is designed to be reusable as a transport container.

6.5.1.2 Secondary Wastes from Truck Shipment of Plutonium

Secondary wastes may be generated in overpack loading and unloading operations at fuel reprocessing plants and at the Federal repository. These wastes are included in the wastes treated at those facilities.

6.5.1.3 Facility Effluents from Truck Shipment of Plutonium

Shipments of plutonium oxide are-subject to radiation dose rate limitations prescribed by the U.S. Department of Transportation (see Section 6.1.1). For shipments in closed vehicles, current Federal regulations impose dose rate limitations of 200 mrem/hr at the external surface of the vehicle, 10 mrem/hr at any point 1.8 m (6 ft) from the vehicle, and 2 mrem/hr at any normally occupied position in the vehicle.

The total rate of heat release from a shipment of ten PuO_2 containers would be about 22 MJ/hr (6 kW), or about the same as that from a truck shipment of spent fuel. Such a heat release would not affect the temperature of the air a few feet from the shipment.

6.5.1.4 Decommissioning of Plutonium Shipping Casks

The useful life of a plutonium shipping cask is estimated to be 20 to 30 years. The level of plutonium contamination is anticipated to be so slight that it is assumed that the cask could be disposed of as non-TRU waste.

6.5.1.5 Postulated Accidents for Truck Shipment of Plutonium

In addition to the use of a Type B overpack, shipments of plutonium oxide would be subject to the packaging requirement of 10 CFR 71.42. This specifies that the inner pressure vessel must not release plutonium when the entire package is subjected to the accident test conditions of 10 CFR 71, Appendix B. Plutonium shipments are also subject to special procedural controls set forth in 10 CFR 73. Transportation accidents similar to those described in Section 6.2.2 (Truck Transport of Spent Fuel) are possible. However, because of the unique precautions required for plutonium shipments, the probability of transportation accidents would be substantially reduced relative to normal truck transport accidents. Furthermore, because of the container design, none of the truck transportation accidents postulated in Section 6.2.2 would result in release of plutonium.

6.5.1.6 Cost of Truck Shipment of Plutonium

Estimates have been made, in mid-1976 dollars, of capital, freight, and total unit costs for truck shipment of plutonium oxide. This section discusses those costs.

Capital Cost of Plutonium Oxide Cask System

The capital cost of the ten casks (overpacks) for one truck shipment of plutonium oxide is estimated to be $220,000, with an accuracy range of ±40%. In making this estimate the PPP-1 casks are treated as purchased equipment supplied repetitively by qualified vendors on a competitive basis. The estimate reflects manufacturers' profits and development, engineering, sales, overhead, and other similar expenses in addition to material and manufacturing costs.

The capital cost estimate includes the cost of 10 secondary pressure vessels (overpacks). It does not include the cost of the transporter, the truck tractor, or the primary pressure vessels.

The accuracy range reflects design uncertainties and the spread in pricing normally experienced in quotes from manufacturers of specialized heavy equipment.

Cask Use Charge for Plutonium Cask

A cask use charge was calculated using the capital cost of the PPP-1 cask overpack and secondary pressure vessel. The primary pressure vessel and canisters containing the PuO_2 are not reusable and were not considered part of the transportation system. The costs of these latter containers are included in the PuO_2 storage facility costs.

The overpack and secondary pressure vessel are assumed to have a 90% use factor (329 days/year) and to be privately owned. For the above basis the daily cask use charge for a single cask is $17/day or $0.53/kg PuO_2-day (equivalent to $5/MTHM-day). Table 6.5.3 shows the cask-day calculations for PPP-1 cask shipments. Multiplying the total cask days by the daily cask use charge gives the unit cask use charge of $32/kg PuO_2 or $0.30/kg HM.

TABLE 6.5.3. Calculations of Cask-Days per Trip for PuO_2 Shipments

Waste Shipment	Distance, miles	Casks Carried/ Trip	Round-Trip Travel Time/ Cask, days	Round-Trip Turnaround Time/Cask, days	Cask-days/ Trip
PuO_2 (PPP-I)	1500	10	4	2	60

Unit Cost Estimate for Truck Transport of Plutonium Oxide

Table 6.5.4 details the haulage charges for truck transport of plutonium oxide. The haulage charge assumes shipment by commercial carrier in specially designed armored vehicles with security-cleared drivers, armed guards, and two armed escort vehicles. Table 6.5.5

TABLE 6.5.4. Haulage Charges for Truck Shipment of Plutonium Oxide[5]

Waste Type	Round-trip Distance, miles	Basic Fee,[a] $/mile	Overcharges,[b] $/mile	Security Procedures,[c] $/mile	Total $/mile	Total $/Trip
PuO_2	3,000	3.18	0.70	0.75	4.63	13,900

a. Includes a specially designed armored vehicle and two armed escort vehicles.
b. Based on a PuO_2 value of $25/g and a charge of $0.03/$100 of valuation over $1,000,000.
c. Includes protective signature service at $15/trip, 2 security-cleared drivers at $0.30/mile + $200/trip, and 2 armed drivers at $0.40/mile.

TABLE 6.5.5. Unit Cost Estimate for Truck Transport of PuO_2 to Independent Storage

Distance, miles	Cask Capacity Equivalent MTHM[a]	Cask Capacity kg PuO_2	Casks/ Trip	Daily Cask Use Charge, $/MTHM/day	Cask Days/Trip	Cask Use Charge/ Trip, $/kg HM	Haulage Charge/ Trip, $/kg HM	Total Unit Cost, $/kg HM	$/g Pu
1500	3.45	32	10	5	60	0.30	0.40 ±40%	0.70	7.6

a. MT of processed spent fuel

shows the unit cost estimate for truck transport of PuO_2 in PPP-1 shipping containers. The unit cost is the sum of the unit cask use charge and the haulage charge.

6.5.1.7 Construction Requirements of Plutonium Shipping Casks

An estimate has been made of the approximate quantities of basic construction materials employed in fabricating 10 casks for shipping plutonium (the number of casks transported on one truck shipment of plutonium oxide). The estimate is given in Table 6.5.6. Quantities listed in the table do not include allowances for primary pressure vessels or canisters, which are disposable items, or for the shielded van, support systems, or other appurtenances.

TABLE 6.5.6. Plutonium Shipping Cask Construction Materials, 10 Casks

Material	Quantity
Steel	5000 kg
Lead	5000 kg
Other	5000 kg

6.5.1.8 Effects of Fuel Cycle Options on Truck Shipment of Plutonium

The reference process for truck shipment of plutonium assumes uranium recycle only with plutonium sent to a Federal repository. The following fuel cycle options have also been assessed insofar as they relate to plutonium transport.

No Recycle

Eliminating the fuel reprocessing operation does away with the generation of separated plutonium. Accordingly, no plutonium transportation is required.

Uranium Recycle Only, with Plutonium to High-Level Waste (HLW)

This alternative would include plutonium with the solidified high-level waste in cask shipments to a Federal repository. No shipment of plutonium alone is required.

Recycle of Uranium and Plutonium

For this fuel cycle option, plutonium would be shipped to a mixed-oxide fuel fabrication plant rather than treated as a waste. This shipment is thus outside the scope of this report. However, the plutonium containers and transport equipment described here are patterned after those developed for use in the uranium-plutonium recycle option.

6.5.2 Rail Transport of Plutonium

Current shipping practice is to transport plutonium by sole-use truck, and it is assumed for planning purposes that this will continue to be the primary shipping mode. However, transport by rail is also possible using the plutonium shipping cask described in Section 6.5.1.1. Shipment of plutonium by rail would also be subject to the physical safeguards requirements of 10 CFR 73. To implement these safeguards requirements for rail shipments of plutonium, it might be necessary to employ special trains.

6.5.3 Physical Protection and Safeguard Requirements for Transport of Plutonium

Plutonium oxide is an attractive target for theft, and it would be a more accessible target during transport than at other stages of the fuel cycle. Sabotage or theft for the purpose of dispersing or threatening dispersal of plutonium oxide, possibly in an extortion threat, may be attractive to a terrorist because of the public perception that plutonium is an extremely poisonous material. It is necessary, therefore, to guard against this possibility even though the biological hazard of plutonium oxide is comparable to that of several other common commercial substances.[6]

Plutonium shipments are safeguarded according to the requirements of 10 CFR 73 and additional guidelines are given in regulatory guides 5.17, 5.31, 5.32 and 5.57.[a] Proposed amendments to 10 CFR 73 would upgrade the safeguarding requirements by defining the performance capabilities required and specifying additional systems and procedures related to planning and scheduling of shipments, security organizational structure, contingency and response plans and procedures, personnel access controls, and security requirements at transfer points.[b] The currently applicable regulations require that shipments of strategic quantities of SNM, e.g., greater than 2000 g of plutonium, be protected from sabotage and theft by the combination of the following procedures and equipment features:

- use of only preplanned routes and schedules, selected to minimize accident and safeguards risks
- hand-to-hand receipts (i.e., signed receipt acceptance of responsibility) at origin, destination and transfers enroute
- no transfers enroute in the case of truck transport
- tamper-indicating seals on containers and locks on containers, if the cargo compartment is not locked
- use of a specially designed truck with immobilizing features (optionally, the drivers may be armed) to deter movement of the vehicle by hijackers, and a penetration resistant cargo compartment that would significantly delay removing the cargo or, accompanying escort car with two armed guards[c]
- continuous radiotelephone communication capability between the truck or rail vehicle, the escort vehicle and the licensee during transport
- call-in requirements at specified maximum intervals during transport
- approved contingency plans for response actions in the event of an attempted hijack or sabotage.

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a. Available from the Nuclear Regulatory Commission, Washington, DC 20555. Attn: Director of Document Control.

b. Federal Register, Vol. 42, No. 128, 34310-34890, July 5, 1977 and Vol. 43, No. 101, 22216-22220, May 24, 1978.

c. 10 CFR 73.31(c) requires one or the other, but not both.

Disabling features in use or under development[7] include armored cabs and cargo compartments, special hinges and locking devices on cargo compartment doors, cargo compartment intrusion penalties (for example, toxic gases or liquids), and such vehicle immobilization features as wheel or steering locks and axle or drive line failures. The special communications and the vehicle disabling features used with truck shipment of plutonium are designed to ensure that an attempted theft or diversion must be carried out with a significant armed attack, that any attempt would be promptly detected, and that a response by additional armed guards or police would be timely and adequate.

The physical protection measures outlined in the previous paragraph are designed to provide reasonable assurance that the plutonium would not be stolen in amounts sufficient for an explosive device or for a contamination that would be significant to the public health and safety. In addition, the close monitoring of shipments would ensure that pursuit would be initiated promptly.

Sabotage of a shipment may immobilize and delay the shipment, but penetration of the cargo compartment and a plutonium oxide shipping container is not likely prior to the time a response force arrives. The shipping cask and cargo container designs provide considerable resistance to penetration or entry with tools, torches or explosives. If a container is ruptured, i.e., by projectiles or explosives, some plutonium oxide contamination of the vehicle and immediate vicinity could occur. However, no hazard to the general public from acts of sabotage to plutonium shipments is expected.

REFERENCES FOR SECTION 6.5

1. Title 10, Code of Federal Regulations, Part 71.42 (10 CFR 71.42).

2. Title 10, Code of Federal Regulations, Part 73 (10 CFR 73).

3. Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle. ERDA-76-43, p. 22.17, Energy Research and Development Administration, Washington, DC, May 1976.

4. R. E. Best and J. L. Ridihalgh, "The Development, Design and Evaluation of a Packaging for the Transportation of Plutonium." Proceedings of the Fourth International Symposium on the Packaging and Transportation of Radioactive Materials, CONF-740901, Miami Beach, FL, September 1974.

5. Tri-State Motor Transit Co., Local Commodity Tariff No. 104-E, Joplin, MO, March 23, 1977.

6. Benard L. Cohen, The Hazards in Plutonium Dispersal. Institute for Energy Analyses, Oak Ridge, Tennessee (Uni. of Pittsburgh, Pittsburgh, PA), March 1975.

7. R. E. Reed, "Design Concepts Study of a Special Nuclear Material Cargo Vehicle." Nuclear Technology, 23:112, August 1974.

6.6 TRANSPORTATION OF NON-HIGH-LEVEL TRU WASTES

6.6 TRANSPORTATION OF NON-HIGH-LEVEL TRU WASTES

Transuranic (TRU) contaminated non-high-level wastes originate from operations in fuel reprocessing plants and mixed-oxide fuel fabrication plants. As prepared for shipment, these wastes are in various forms which may be generally classified in three categories:

1. combustible trash
2. wet wastes, particulate solids, incinerator ash, and filter media immobilized by incorporation with a solidifying agent
3. failed equipment and noncombustible waste.

TRU-contaminated wet wastes are assumed to be solidified prior to shipment. Combustible materials may be either packaged without treatment or incinerated and the ash immobilized by incorporation in a solidifying agent.

Under a proposed rule,[1] commercially generated wastes contaminated with greater than 10 nanocuries of transuranic elements per gram of waste (TRU waste) would be sent to Federal repositories for interim storage or permanent isolation. Individual waste packages from categories 2 and 3 would likely exceed the 0.001 Ci limitation for Group I radionuclides (see Sections 6.1.1.1 and 6.1.1.4) and would require shipment in overpacks that meet Type B package standards. In many instances, incineration of combustible trash and compaction of non-combustible trash would increase TRU concentrations for these latter wastes so as to require shipment in Type B overpacks. While it is likely that some TRU-waste packages from fuel reprocessing plants and mixed-oxide fuel fabrication plants could be shipped in Type A packaging, it seems probable that most of these shipments would require Type B packaging. For purposes of this study, all non-high-level TRU waste shipments from fuel reprocessing plants and mixed oxide fuel fabrication plants are assumed to require shipment in overpacks that meet Type B package standards or their equivalent.

At present there is no commercial reprocessing of spent fuel in the U.S., and thus no commercial shipments of non-high-level TRU waste are being made. However, TRU waste is routinely shipped from Department of Energy (DOE) contractor sites to an interim storage facility located at the Idaho National Engineering Laboratory (INEL). Disposable containers currently in use by DOE contractors include 55-gal steel drums that incorporate plastic liners, and plywood boxes coated with fiberglass-reinforced plastic.[2] The drums and boxes are loaded into steel cargo containers* 2.4 x 2.4 x 6.1 m (8 x 8 x 20 ft), and the cargo containers are transported by special permit arrangement in ATMX-series rail cars.[3,4] Wastes for storage at INEL must be packaged in a manner permitting retrievability after interim storage of up to 20 years. It is assumed that a retrievability packaging requirement would also be in effect for commercial wastes shipped to interim storage prior to final isolation at a Federal repository.

6.6.1 Truck Transport of Non-High-Level TRU Wastes

Shipments of non-high-level TRU wastes could be made by truck or rail. For planning, it is assumed that the majority of these shipments will be by truck.

*These cargo containers are for convenience and are not a safety requirement

Table 6.6.1 provides example shipping information for shipments of non-high-level TRU wastes from fuel reprocessing plants and mixed oxide fuel fabrication plants to Federal repositories. The table is based on volumes of non-high-level TRU wastes projected for the year 2000 (see Section 2.1) and on assumptions about shipping modes described in Section 6.6.1.1. Only shipment projections for the reference waste treatment option (incineration of combustible waste with cementation of ash and wet wastes) are presented. Other treatment options would significantly change the total number of waste containers required for shipment. For example, minimum treatment of the waste with either cementation or bitumenization of wet wastes would increase the number of waste drums in the <0.2 R/hr and 0.2-1.0 R/hr surface activity categories by about a factor of three while reducing slightly the number of waste drums in the 1.0-10.0 R/hr and >10.0 R/hr surface activity categories.

TABLE 6.6.1. Information on Truck Shipment of Non-High-Level TRU Wastes in the Year 2000[a]

	No. of Containers by Surface Activity, R/hr			
Container Type	<0.2	0.2-1.0	1.0-10.0	>10.0
55 gal drums	20,600	7,300	7,400	15,900
canister, 0.76 m dia		251	15	
box, 1.2 x 1.8 x 1.8 m	324			

No. of Shipments by Shipping Mode				
Unshielded Van[b]	Shielded Van[c]	Cask w/2 1/2-in. Lead Equivalent Shield[d]	Cask w/4 1/2-in. Lead Equivalent Shield[e]	Canister Cask[f]
680	203	529	2,650	87

a. Treatment option includes incineration of combustible waste with cementation of ash and wet wastes.
b. Assumed capacity 36 drums or 3 boxes.
c. Assumed capacity 36 drums or 3 boxes.
d. Assumed capacity 14 drums.
e. Assumed capacity 6 drums.
f. Canister cask described in Section 6.4.1. Shipment will be by rail.

6.6.1.1 Containers for Truck Transport of Non-High-Level TRU Wastes

Prior to shipment, non-high-level TRU wastes are packaged in disposable containers. The waste containers are assumed to be transported in Type B overpacks which may be unshielded or shielded depending on the radiation level at the surface of a disposable container.

Disposable containers are assumed to include DOT specification 17C 55-gal steel drums[5,6] and DOT specification 7A steel boxes[7] having dimensions of 1.2 x 1.8 x 1.8 m (4 x 6 x 6 ft). This box volume is about the same as that occupied by a palletized array of 55-gal drums (two drums by three drums) on two levels, for a total of 12 drums per pallet.

Other disposable containers may also be used. For example, because of shielding requirements, some shipments of TRU-contaminated equipment and metal scrap are assumed to be packaged in the canister described for containment of fuel residues. Shipment would probably be made by

rail in the fuel residues cask described in Section 6.4.1. HEPA filters which are too large to be accommodated in 55-gal drums are assumed to be packaged in 80-gal drums.*

To satisfy the requirements for retrievability, if shipments are made to interim storage rather than to a permanent disposal facility, it may be necessary to provide either a liner or a disposable overpack for drums and boxes.

All TRU waste packaged in disposable containers will be shipped in overpacks that meet Type B package requirements. The mode of shipment will depend on the radiation level at the surface of the disposable container. Regulations permit disposable containers with surface dose rates less than 200 mR/hr to be shipped without shielding. Most 55-gal drums in this category will actually have surface dose rates less than 1 mR/hr. Lead liners can be added to 55-gal drums to reduce surface dose rates to acceptable limits if required for the protection of personnel during handling.

For planning purposes, four shipment modes have been defined for TRU wastes packaged in 55-gal drums. Table 6.6.2 describes these shipment modes.

TABLE 6.6.2. Shipment Modes for 55-Gal Drums of Non-High-Level TRU Waste

Drum Surface Dose Rate	Shipment Mode[a]
<200 mR/hr	Unshielded sole-use van
200 mR/hr-1 R/hr	Shielded sole-use van
1 R/hr-10 R/hr	Cask with equivalent shield thickness of 5 cm lead + 2 cm steel
>10 R/hr	Cask with equivalent shield thickness of 10 cm lead +2.5 cm steel

a. All overpacks meet Type B package requirements.

Reusable transport shields constructed to Type B package standards are available both for shipments that do not require shielding and for those that do require shielding. Some commercially available units are listed in Table 6.6.3. Shielded Type B overpacks are typically smaller than unshielded units. The container volume in reusable transport shields must be limited so that shielded weight does not cause overall package weights to exceed truck weight limits.

Drums and boxes that do not require shielding are assumed to be transported in a Super Tiger®.[8] The Super Tiger® is a double-walled steel box with a fire-resistant polyurethane foam filler for shock and thermal insulation. Interior dimensions are 1.93 x 1.93 x 4.36 m (76 x 76 x 172 in.). The empty weight is 6,800 kg (15,000 lb), and the maximum payload is 13,600 kg (30,000 lb). Figure 6.6.1 is a schematic of the container. Three pallets each containing twelve 55-gal drums (total of 36 drums) or three steel boxes could be transported in a Super Tiger®. The overpack is loaded from one end in a horizontal position.

* Eighty-gallon drums shall meet the requirements of MS-27683-22 and shall meet or exceed the requirements of 49 CFR 173.395a.

TABLE 6.6.3. Reusable Type B Transport Shields for 55-Gal Drums

Cask Identification	Cask Type	Equivalent Shield Thickness, cm	Capacity No. of Drums	Capacity m^3	Empty Weight, kg	Company
HN-200	Top load	7.6 lead + 3.2 steel	3	2.1	16,600	Hittman
BS-33-180	Top load	8.9 steel	8	5.1	15,000	ATCOR
LL-50-100	Top load	11.4 lead	8	2.8	22,700	ATCOR
BC-48-220	Top load	17.8 concrete + 6.4 steel	14	6.2	21,800	ATCOR
S3-205	Top load	7.6 steel	15	9.1	19,100	NECO
Super Tiger	End load	1.3 steel	42	16.3	6,800	NECO

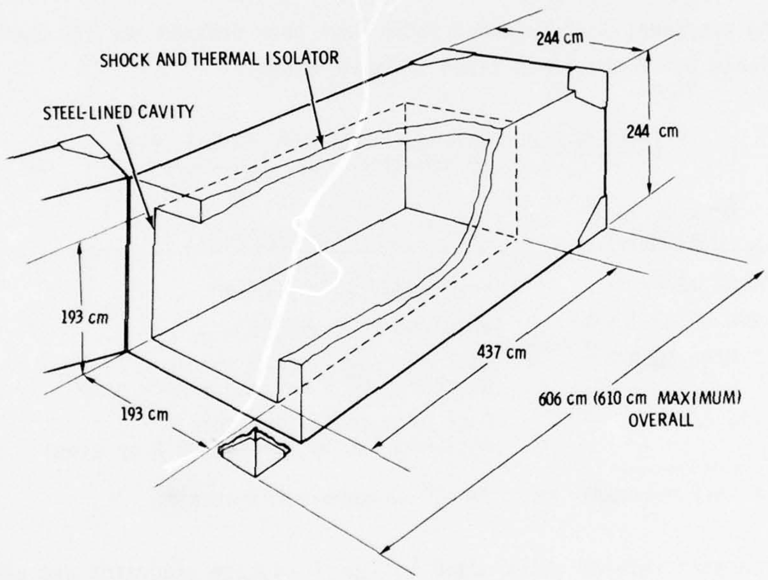

FIGURE 6.6.1. Super Tiger

Shielded vans licensed for Type B shipments are not available commercially although shielded vans for LSA shipments of drummed waste are in commercial use (for example, the ATCOR and Hittman shielded vans). It is anticipated that a shielded van which meets Type B package standards or a Super Tiger-type overpack which incorporates some shielding could be built to transport drummed TRU waste with surface dose rates in the 1 R/hr range.

For planning purposes it is assumed that a Type B transport shield incorporating about 6 cm of lead equivalent shielding would transport 14 drums and that a shield incorporating about 12 cm of lead equivalent shielding would transport six drums of TRU waste.

6.6.1.2 Secondary Wastes from Truck Shipment of Non-High-Level TRU Waste

Secondary wastes may be generated in loading and unloading operations at fuel reprocessing and mixed-oxide fuel fabrication plants and at Federal facilities for interim storage or final isolation. These wastes are included in the discussions on secondary wastes under each treatment facility.

6.6.1.3 Emissions from Truck Shipment of Non-High-Level TRU Waste

Shipments of non-high-level TRU waste would be subject to radiation dose rate limitations prescribed by the U.S. Department of Transportation (see Section 6.6.1). For shipments in closed vehicles, current Federal regulations impose dose rate restrictions of 200 mrem/hr at the external surface of the vehicle, 10 mrem/hr at any point 6 ft from the vehicle, and 2 mrem/hr at any normally occupied position in the vehicle.

Heat generation rates from routine shipment of non-high-level TRU waste would be so low as to have negligible environmental effects.

6.6.1.4 Decommissioning of Containers for Truck Shipment of Non-High-Level TRU Waste

Non-high-level TRU waste will be packaged in disposable containers and shipped in reusable Type B transport shields. The Type B transport shields are expected to have a useful life of 20 to 30 years. Decommissioning will be accomplished by appropriate decontamination procedures followed by disposal as non-TRU waste.

6.6.1.5 Postulated Accidents for Truck Shipment of Non-High-Level TRU Waste

Non-high-level TRU waste will be packaged in steel drums and steel boxes and shipped in overpacks which meet Type B package standards. The accident protection provided by these overpacks will enable a shipment to withstand all but very severe, highly unusual accidents. Only the most severe impact and longest fire is assumed to provide a mechanism such that some radioactivity might be lost. The material being transported is solid and, for the most part, noncombustible; hence the fraction of respirable material released in a severe accident which ruptures a shipping container is expected to be small.

Postulated minor and severe accident scenarios for truck shipment of non-high-level TRU waste are given in Tables 6.6.4 and 6.6.5. Expected frequencies of postulated accidents are based on accident probabilities per vehicle mile from Section 6.1.3.

For purposes of environmental consequence analysis, the material releases associated with accidents numbers 6.6.3 and 6.6.4 in Tables 6.6.4 and 6.6.5 have been selected as umbrella source terms. (The concept of an umbrella source term is explained in Section 3.7.) This means that the releases from these accidents are the largest in their respective source term categories. The environmental consequences of these accidents are described in DOE/ET/0029. Accidents are cross indexed with their appropriate umbrella source term in Appendix A, Section 3.

TABLE 6.6.4. Non-High-Level TRU Waste Containers - Minor Accidents

Accident No. and Description	Sequence of Events	Safety System	Release
6.6.1 — Truck collision or overturn accident involves non-high-level TRU waste container. Expected frequency ~3 per year.	1. Collision or overturn accident occurs. 2. Truck leaves roadway and may overturn. 3. Confinement barriers of Type B overpack remain intact. 4. Accident is reported to local and Federal officials. 5. Package recovered.	1. Radiation warning signs on overpack caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site and recover overpack. 3. Confinement barriers of Type B overpack contain all radioactive material.	None
6.6.2 — Truck collision or overturn accident and 1/2 hour (or less) fire involves non-high-level TRU waste container. Expected frequency ~0.3 per year.	1. Collision or overturn accident occurs. 2. Truck leaves roadway and may overturn. 3. Non-high-level TRU waste container is involved in 1/2 hour (or less) fire. 5. Accident is reported to local and Federal officials. 6. Package recovered.	1. Radiation warning signs on overpack caution onlookers to keep distance. 2. Interagency radiological assistance personnel available to assist local public safety and transport carrier officials to control site and recover overpack. 3. Confinement barriers of Type B overpack contain all radioactive material.	None
6.6.3 — Non-high-level TRU waste shipment made in improperly closed packages. Expected frequency ~1 per year.	1. A shipment of non-high-level TRU waste is made with improperly closed packages. 2. Some radioactive material is released from the packages to the interior of the overpack. 3. A small fraction of the material released from the waste packages escapes from the confinement of the overpack during the trip.	1. Radiation warning signs on overpack caution onlookers to keep distance. 2. Quality assurance and package test requirements reduce probability of shipments with improperly closed packages. 3. Sealed overpack prevents release to the environment of material spilled from improperly closed packages.	Release fraction is 10^{-8} of shipment inventory. Assume release of respirable out off-gas system control.

TABLE 6.6.5. Non-High-Level TRU Waste Containers Severe Accidents

Accident No. and Description	Sequence of Events	Safety System	Release
6.6.4 — Non-high-level TRU waste container is subjected to severe impact and fire. Expected frequency ~3 x 10^{-5} per year.	1. Collision or overturn accident occurs at high speed. 2. Overpack strikes massive object and decelerates almost instantaneously. 3. Overpack is involved in fire which *lasts longer than 1 hour.* 4. Waste packages are breached by force of accident. 5. Only rips or holes are created in overpack; waste packages are retained within the confines of the overpack. 6. Accident is reported to local and Federal officials. 7. Package sealed. 8. Area decontaminated. 9. Package recovered.	1. Interagency radiological assistance to personnel available to assist local public safety and transport carrier officials to control site and recover overpack. 2. Only small openings exist in *overpack. Accident does not* result in gross breach of containment.	Release fraction is 10^{-5} of shipment inventory. Assume release of respirable particles at ground level without off-gas system control. See Table 5.3.1 for contents of packages and Table 6.6.1 and 6.6.2 for the number of packages per shipment.

6.6.1.6 Cost of Truck Shipment of Non-High-Level TRU Wastes

Estimates have been made, in mid-1976 dollars, of capital, operating, and levelized unit costs. These costs are discussed in this section.

Capital Cost of Non-High-Level TRU Waste Cask Systems

The capital cost of a system for truck transport of drums of TRU waste which do not require shielding is estimated to be \$85,000 with an accuracy range of ±40%. The system consists of an unshielded Type B overpack with capacity for thirty-six 55-gal drums on three pallets or an equivalent volume of boxes, a special low-boy trailer, and appropriate tiedown equipment. Costs of the disposable waste containers and of the truck tractor are excluded from the estimate. The capital cost of a 36-drum shielded Type B overpack is estimated to be \$135,000 ±40%.

The capital cost of a system for truck transport of drums of TRU waste which are shipped in reusable Type B shielded casks is estimated to be \$120,000 ±40% for a cask system designed to transport 14 drums with a 1 to 10 R/hr surface dose rate, and \$150,000 ±40% for a cask system designed to transport 6 drums with a surface dose rate >10 R/hr. Costs include allowances for special low-boy trailers to transport the casks and for appropriate tiedown equipment. Costs of the disposable waste containers and of truck tractors are excluded from the estimates.

The accuracy range reflects the uncertainties in the optimum design of systems for transportation of non-high-level TRU waste. Additionally, the range reflects the spread in pricing normally experienced in quotes from manufacturers of specialized heavy equipment.

Cask Use Charges for Non-High-Level TRU Wastes

Four different systems are required, depending on the surface dose rate of the waste containers. These systems and their related cask use charges are listed below.

Containers with <0.2 R/hr Surface Radiation. Containers in this category will travel in a Super Tiger, Type B shipping container having a 36-drum capacity. The cask use charge for the container, assuming a 90% use factor (329 days/year) and private ownership, is \$70/day.

Containers with 0.2 - 1.0 R/hr Surface Radiation. Containers in this category will travel in a 36-drum shielded van. The use charge for this vehicle, assuming a 90% use factor and private ownership, is \$110/day.

Containers with 1.0 - 10 R/hr Surface Radiation. Containers in this category will travel in a 14-drum shielded cask. The cask use charge, assuming a 90% use factor and private ownership, is \$90/day.

Containers >10 R/hr Surface Radiation. Containers in this category will travel in a six-drum heavily shielded cask. The cask use charge, assuming a 90% use factor and private ownership, is \$120/day.

Unit Cost Estimate for Truck Transport of Non-High-Level TRU Wastes

The unit cost estimate consists of the sum of the unit cask-use charge and the unit haulage charge. Haulage charge calculations are shown in Table 6.6.6. Table 6.6.7 shows the derivation of the total unit cost estimate for truck transport of non-high-level TRU waste from a fuel

TABLE 6.6.6. Haulage Charges for Truck Shipment of Non-High-Level TRU Waste[9]

Waste Type	Distance, miles(a)	Basic Fee, $/mile	Decontamination Holdover, $/mile(b)	Second Driver, $/mile	Total $/mile	Total $/Trip
NHL TRU waste						
<0.2 R/hr	1500*	0.98	0.01	0.15	1.14	1700
0.2-1.0 R/hr	3000	0.91	0.01	0.15	1.07	3200
1.0-10 R/hr	3000	0.91	0.01	0.15	1.07	3200
>10 R/hr	3000	0.91	0.01	0.15	1.07	3200

a. Round-trip except for * distance, which is one-way.
b. Based on an average holdover time of two days at $10.50/day.

TABLE 6.6.7. *Unit Cost Estimate for Truck Transport of TRU Waste from FRP to Federal Repository in Terms of Cost per Container*

Surface Radiation, R/hr	Distance, miles	Cask Capacity: 55-gal Drum	Cask Capacity: 80-gal Drum	Cask Capacity: Boxes	Casks per Trip	Daily Casks-Use Charge, $/cask/day	Cask-days/ Trip	Total Cask Use Charge, $/cask/Trip	Haulage Charge, $/Trip	Total Unit Cost, $ per Container(a): Cask	55-gal Drum	80-gal Drum	Box
< 0.2	1500	36	-	3	1	70	7	500	1700	2200	60	-	760
0.2-1	1500	36	-	-	1	110	7	800	3200	4000	110	-	-
1-10	1500	14	-	-	1	90	7	600	3200	3800	270	-	-
>10	1500	6	6(b)	-	1	120	7	850	3200	4050	670	670	-

a. Some TRU waste having surface radiation in the range of 0.2-10 R/hr will be shipped by rail in 30-in-dia canisters similar to fuel residue canisters. The shipping cost is assumed to be the same as for a canister of fuel residue, or $12,500 per canister.
b. The 6-drum cask is assumed to be sized for 80-gal drums.

reprocessing plant to a Federal repository in terms of cost per container. Assuming the same shipping distance from both MOX-FFP and FRP to a Federal repository, the unit cost of transportation of non-high-level TRU waste from a MOX-FFP to a Federal repository will be the same as that shown in Table 6.6.7 for the 0.2 R/hr waste shipment. Table 6.6.8 shows unit costs for Transportation of all non-high level waste (excluding fuel residue) from the FRP and MOX-FFP expressed in terms of cost per kilogram of heavy metal processed. Because the amount of waste shipped varies with the Treatment process, the unit costs are shown for the four possible combinations of Treatment processes discussed in this document. These costs range from $1.60 to $4.20 per kilogram of heavy metal.

6.6.1.7 Construction Impacts of Transport Shields for Truck Shipment of Non-High-Level TRU Wastes

Estimates have been made of the approximate quantities of basic construction materials employed in fabricating reusable transport shields for truck shipment of non-high-level TRU wastes. The estimates are given in Table 6.6.9. Quantities listed in the table do not include allowances for disposable containers, low-boy trailers, support systems, or other appurtenances.

6.6.1.8 Effects of Fuel Cycle Options on Truck Shipment of Non-High-Level TRU Wastes

The reference process for truck shipment of non-high-level TRU waste assumes reprocessing of spent LWR fuel and recycling the retrieved uranium and plutonium. The following alternative fuel cycle modes have also been assessed insofar as they relate to transport of non-high-level TRU wastes.

No Recycle

For the no recycle option, non-high-level TRU wastes are not generated and there is no transportation requirement.

Uranium Recycle Only

Mixed-oxide fuel would not be fabricated in this fuel cycle option, and non-high-level TRU waste shipments would not originate at MOX plants. The number of waste shipments shown in Table 6.6.1 would be reduced to those originating at fuel reprocessing plants. Assuming the same reprocessing rate with or without plutonium recycle, the number of waste shipments originating at fuel reprocessing plants in the year 2000 is shown in Table 6.6.10. Since all waste shipments from MOX plants are in containers with surface dose rates less than 0.2 R/hr, the reduction in the transportation requirement for the uranium recycle only option occurs solely in the number of unshielded van shipments.

6.6.2 Rail Transport of Non-High-Level TRU Wastes

Reusable transport shields used for truck transport could also be used for rail shipment of non-high-level TRU waste. The outside dimensions of the Super Tiger are about the same as those of a standard steel cargo container, 2.4 x 2.4 x 6.1 m (8 x 8 x 20 ft). Two Super Tiger containers could be transported in an ATMX-600 rail car. Casks for shipment of non-high-level TRU wastes are smaller than spent fuel rail casks, and two or more TRU waste casks could be transported on a single railroad flatcar.

TABLE 6.6.8. Unit Cost Estimate for Transport of Reference System NHL TRU Waste to Federal Repository in Terms of Cost per kg HM Processed[a,b]

Waste Immobilization Process	Surface Radiation Rate, R/hr	No. of TRU Waste Containers/MTHM: 55-gal Drums	80-gal Drums	Boxes	30-in.-dia Canisters	Unit Cost, $/kg HM: 55-gal Drums	80-gal Drums	Boxes	30-in.-dia Canisters	Total
Incineration with cementation of ash and wet wastes (reference process)	<0.2	2.432	-	0.035	-	0.15	-	0.03	-	0.18
	0.2-1	0.976	-	-	0.034	0.11	-	-	0.43	0.54
	1.0-10	0.983	-	-	0.002	0.27	-	-	0.03	0.30
	>10	2.129	-	-	-	1.43	-	-	-	1.43
Total						1.96		0.03	0.46	2.45
Incineration with bituminization of ash and wet wastes	<0.2	1.338	-	0.035	-	0.08	-	0.03	-	0.11
	0.2-1	1.728	-	-	0.034	0.19	-	-	0.43	0.62
	1-10	0.983	-	-	0.002	0.27	-	-	0.03	0.30
	>10	0.943	-	-	-	0.63	-	-	-	0.63
Total						1.17		0.03	0.46	1.66
Minimum treatment with cementation of wet wastes	<0.2	7.317	-	0.035	-	0.44	-	0.03	-	0.47
	0.2-1	1.336	-	-	0.034	0.15	-	-	0.43	0.58
	1-10	4.223	-	-	0.002	1.14	-	-	0.03	1.17
	>10	1.551	1.450	-	-	1.04	0.97	-	0	2.01
Total						2.77	0.97	0.03	0.46	4.23
Minimum treatment with bituminization of wet wastes	<0.2	7.338	-	0.035	-	0.44	-	0.03	-	0.47
	0.2-1	1.666	-	-	0.034	0.18	-	-	0.43	0.61
	1-10	4.223	-	-	0.002	1.14	-	-	0.03	1.17
	>10	0.615	1.450	-	-	0.41	0.97	-	-	1.38
Total						2.17	0.97	0.03	0.46	3.63

a. The cost uncertainty for these NHL TRU waste shipments is ±15% excluding the effects of distance and treatment combinations.

b. Cost of fuel residue shipments not included. See Section 6.4.

TABLE 6.6.9. Construction Materials for Reusable Type B Transport Shields

Container	Material	Quantity, kg
36-drum unshielded overpack	Steel	7
36-drum shielded overpack	Steel	7
	Lead	12
14-drum cask	Steel	5
	Lead	14
6-drum cask	Steel	4
	Lead	15

TABLE 6.6.10. Information on Truck Shipment of Non-High-Level TRU Wastes from Fuel Reprocessing Plants in the Year 2000, Uranium Recycle Only Fuel Cycle Mode[a]

Container	No. of Containers by Surface Activity, R/hr <0.2	0.2-1.0	1-10	>10
Drum - 55-gal	10,300	7,300	7,400	15,900
Canister - 30-in. dia		251	15	
Box - 4 x 6 x 6 ft	225			

Shipping Mode	No. of *Shipments*
Unshielded van[b]	361
Shielded van[c]	203
Cask with 2-1/2 in. lead equivalent shielding[d]	529
Cask with 4-1/2 in. lead equivalent shielding[e]	2,650
Canisters cask[f]	87

a. Reference waste treatment system includes incineration of combustible wastes and immobilization in cement of wet wastes and particulate solids.
b. Assumed capacity is 36 drums or 3 boxes.
c. Assumed capacity is 36 drums.
d. Assumed capacity is 14 drums.
e. Assumed capacity is 6 drums.
f. Canister cask is described in Section 6.4.1.

Current practice in the handling of large volumes of TRU wastes by prime contractors of the Department of Energy is to place 55-gal drums of waste inside a steel cargo carrier, 2.4 x 2.4 x 6.1 m, and ship two cargo carriers in an ATMX-600 rail car. Shipment is on the basis of a special permit. Under present regulations, shipment of commercial wastes in this manner would require the approval of the combination cargo carrier-rail car as a Type B container.

6.6.3 Physical Protection and Safeguard Requirements for Transport of Non-High-Level TRU Waste

The non-high-level waste would be relatively unattractive to an adversary because of its high and varied dilution of radioactive materials. No container or single shipment would contain a strategic quantity of plutonium, and the material, a large share of which would be solidified in concrete or bitumen, would not be a threat to the public as a dispersable radioactive containment. In addition, those drums with dose rates above 10 R/hr would be overpacked in 15-ton biological shields.

No special safeguarding during shipment of these wastes is required. If sabotage of a shipment occurred, the release of radioactive material, even under several conditions, is expected to be small (see 6.6.1.5).

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7.0 FINAL ISOLATION AND DISPOSAL OF LONG-LIVED WASTES

7.0 FINAL ISOLATION AND DISPOSAL OF LONG-LIVED WASTES

This section describes the geologic considerations essential for repository selection, the nature of geologic formations that are potential repository media, the thermal criteria for waste placement in geologic repositories, and conceptual repositories in four different geologic media. More extensive discussions of geologic media properties are available in References 1 and 2, which present results of geologic isolation studies carried out in support of the CWMS.[3] The repository design concepts presented in References 1 and 2 have been modified here to accommodate the waste types and quantities estimated for this generic system. Also, the design basis has been supplemented by additional thermal analysis, design, and cost studies, the results of which are presented here. The repository site selection process, the adequacy of the data base, and site qualification considerations are discussed in the CWMS[3] and are not extensively treated here.

The objective for the final isolation of radioactive waste is to prevent the waste from becoming a threat to human health and safety during its hazardous lifetime. The presence of appreciable concentrated quantities of long-lived radioactive elements in spent fuel or HLW and other TRU wastes will require that these wastes be effectively isolated (contained or released in nonhazardous concentrations to man's environment) for thousands of years. There is a general concensus that we cannot rely alone on the continuity of existing governments and institutions over this long time period to insure isolation of the concentrated wastes. Therefore, isolation of this waste should require no maintenance and only minimal control by existing persons or institutions. Isolation of concentrated radioactive wastes in deep geologic repositories is considered an effective means of permanent waste isolation within the scope of available technology. Alternatives to geologic isolation are discussed extensively in the CWMS[3] and are not described in this report.

At a geologic repository, the requirement of waste isolation is met by 1) placing the waste deep underground, 2) backfilling the emplacement excavations, and 3) filling, plugging and sealing all access shafts from the surface. This type of deep underground placement of the radioactive wastes effectively removes them from all normal surficial processes that could unearth them. Wastes are further isolated by selecting a stable waste form and canister material and by selecting a rock formation and a location with a long history of geologic stability that can be, on the basis of present understanding of geologic processes, extrapolated to remain stable in the future. This concept is referred to as the multibarrier approach to waste isolation.

Options available within the deep geologic repository concept relate primarily to alternative geologic formations and methods of emplacement. Before a specific geologic formation is judged acceptable for waste disposal, it will be evaluated on the basis of parameters described in Section 7.2.1. Geologic formations judged potentially acceptable as waste repository media, for which conceptual repository designs have been prepared for this report, are bedded and domed salt, crystalline rock (granite), argillaceous (shale and clay) formations, and volcanics (basalt). Although repositories might be feasible in other rock such as tuff and dry limestone,

present information is inadequate to discuss such repositories in the same detail. The pertinent characteristics of each rock formation are discussed in Section 7.2.2.

A typical Federal repository would consist of numerous chambers excavated in a suitable geologic formation. Shafts and surface structures would be designed to support the repository operation. Figure 7.1 shows an artist's concept of a waste repository and its support facilities. Wastes would be received at the repository's surface facilities. All waste shipments would arrive in containers that complied with DOT regulations for radioactive shipments (Section 6) and be segregated according to waste type. Containers showing signs of damage or leakage would be overpacked in specially designed containers. Finally, the containerized waste would be lowered into the repository and emplaced in the appropriate chamber. Chambers would be backfilled after emplacement of specified quantities of wastes. After the entire repository had been filled, the surface facilities would be decommissioned and the shafts would be filled and plugged.

REFERENCES FOR SECTION 7.0

1. Union Carbide Corporation, Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Y/OWI/TM-44, Office of Waste Isolation, Union Carbide Corporation, Nuclear Division, Oak Ridge, TN, 1978.

2. Technical Support GEIS: Radioactive Waste Isolation in Geologic Formations, Y/OWI/TM-36, prepared for the Office of Waste Isolation, Union Carbide Corporation, Oak Ridge, TN, 1978.

3. Environmental Impact Statement Management of Commercially Generated Radioactive Waste, DOE/EIS-0046-D, U.S. Department of Energy, in preparation for publication at the time this report went to press.

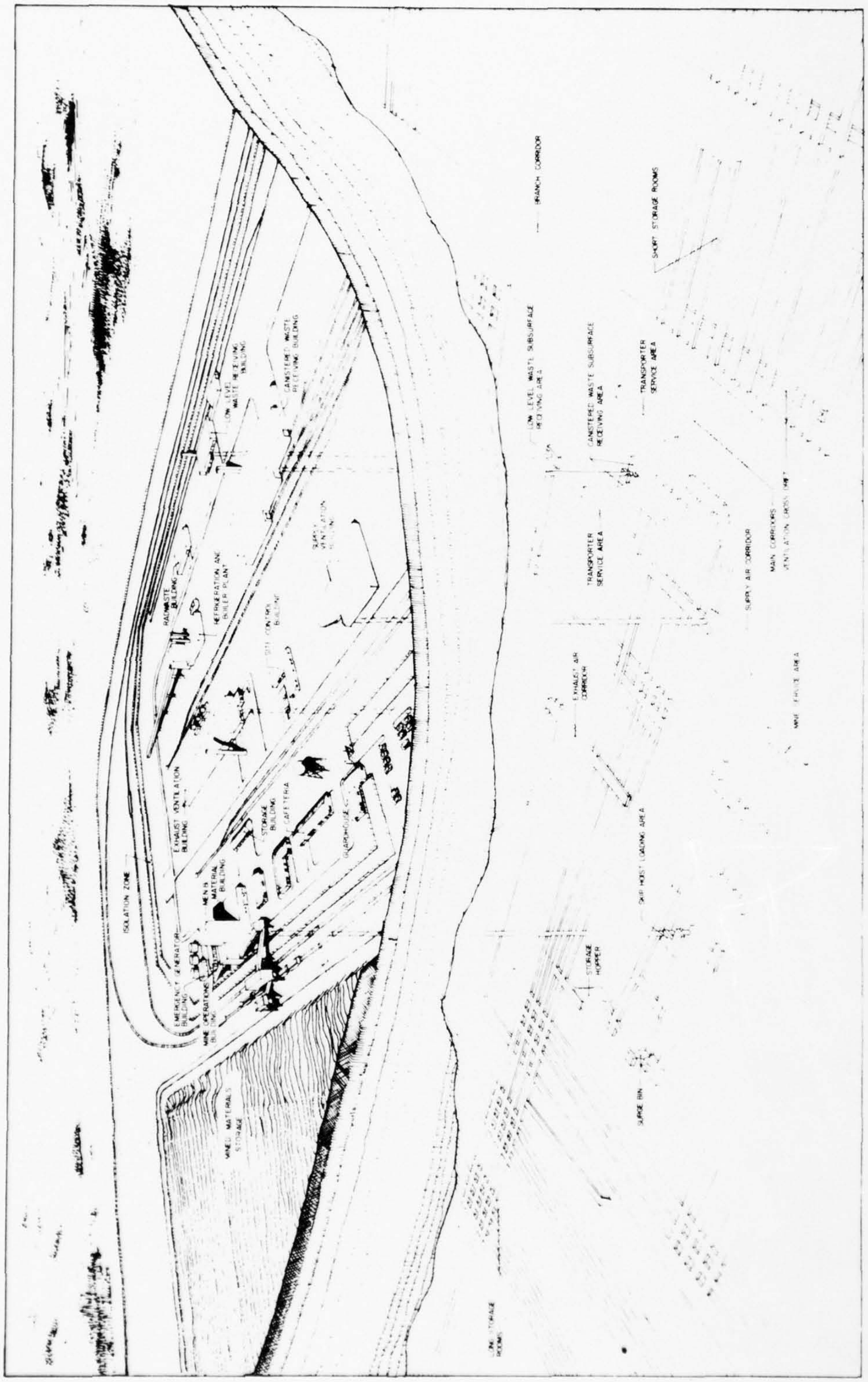

FIGURE 7.1. Waste Repository Perspective

7.1 BACKGROUND

7.1 BACKGROUND

Since its earliest days, the nuclear power program has included the study of techniques for the permanent isolation of radioactive wastes. A major objective has been to ensure isolation of the wastes from the biosphere by methods requiring minimal human surveillance and intervention during the time the waste could be a potential radiological hazard. Included in these studies has been the development of technology for converting liquid or soluble wastes to leach-resistant solids, and final isolation of the solidified wastes. Solidification not only decreases the potential for an accidental release to the environment, but permits safe shipment to a final isolation site, and generally reduces the volumes of waste to be stored.

Locations considered desirable for ultimate isolation are those where the geology would be expected to remain essentially unchanged for thousands of years. Abandoned dry mines, desert sands, deep beds of clay, the sea bed, and locations where continual deposition of fresh material occurs (e.g., alluvial deposits) were considered as potential locations. Isolation of liquid wastes in pumped out (or dry) wells and salt cavities was also examined. Early in the petroleum industry, when surface drainage was becoming contaminated by oil-field brines, operators developed the practice of injecting these saline wastes into permeable subsurface sandstones. The petroleum industry's use of salt formations since 1950 to economically store liquefied petroleum gases prompted the suggestion that radioactive wastes be stored in salt formation cavities.

The Atomic Energy Commission (AEC) first asked the National Academy of Sciences (NAS) in 1954 to help evaluate methods of isolating liquid radioactive wastes underground. A steering committee was formed that sponsored a conference on this topic at Princeton University in September 1955. A permanent committee composed of prominent earth scientists was set up in late 1955 as part of the Academy's earth sciences division. Until 1968 it was known as the Committee on Geologic Aspects of Radioactive Waste Disposal; in 1968 it was renamed the Committee on Radioactive Waste Management (CRWM).

The NAS Committee's first major report, The Disposal of Radioactive Waste on Land, was published in 1957. Specific recommendations on waste isolation were: 1) storage in tanks was at that time the safest and possibly the most economical method of containing waste; 2) isolation of wastes in salt was the most promising method for the near future; and 3) the next most promising method appeared to be stabilization of the waste in a slag or ceramic material forming a relatively insoluble product. (This report also contains a history of the committee and the proceedings of the Princeton Conference as appendices. The Committee on Deep Disposal at the Princeton Conference recommended that "waste be disposed of without concern for its recovery.")[1]

The concept of disposing of solidified radioactive wastes in cavities mined in salt beds or domes was suggested in the late 1950's. With demonstration of solidification processes, the feasibility of storing solidified radioactive waste in buried salt was explored in an experiment known as Project Salt Vault[2] conducted by ORNL in an unused salt mine near Lyons,

Kansas, between 1963 and 1968. In this experiment, containers of highly radioactive test reactor fuel, used to simulate the radioactive and thermal properties of solidified waste, were buried (and later retrieved) and the effects on the salt were measured. This experiment obtained data on the properties of salt at elevated temperatures and indicated that there were no immediate detrimental effects on the stability of the salt as a result of exposure to heat or radiation. It was noted that elevated temperatures would cause accelerated creep (or closure), and this caused some concern about the structural stability of mined areas. However, additional data obtained on characteristics of salt have made it possible to prepare conceptual designs for mined salt disposal facilities. A review in 1970 by the NAS Committee on Radioactive Waste Management endorsed bedded salt formations as the most promising medium for ultimate isolation of radioactive wastes.

Based on the results of the waste solidification program and Project Salt Vault, the AEC proposed use of the abandoned salt mine at Lyons, Kansas, as a pilot facility and an initial Federal repository for disposal of commercial high-level wastes, subject to the satisfactory completion of certain additional tests and studies.

One item on this list of additional studies, and another issue raised later, involved possible routes of entry of water into the Lyons mine and cast doubt on the feasibility of using that particular mine for waste disposal. One factor was that at least 29 abandoned gas and oil boreholes extended into or below the salt formation near the site. While 26 of these boreholes could probably be plugged successfully, the likelihood of plugging the other three was "very low." A second factor was that an adjacent salt mine had made extensive use of "solution mining," in which water is used to dissolve the salt. Such mining leaves no supporting pillars underground, which could lead to collapse of a large area. The situation was further complicated by the fact that at the adjacent salt mine a hydraulic fracturing technique was used, in which water is forced down one hole where it cracks the salt bed, works its way over to a second hole, and then returns to the surface carrying dissolved salt. This technique worked well for a brief period in the mid-1960s, until some 680,000 ℓ (180,000 gal) of water disappeared. Because of uncertainties arising from these factors, the AEC decided to terminate work at Lyons, and the site was returned to its owners in contamination-free condition.

After the work at Lyons was stopped, other potential Kansas sites were studied for suitability for a waste repository. Studies were also expanded to evaluate the suitability of all potential geologic formations and rock types in the continental United States for waste repositories. Geologic formations under investigation included: 1) rock salt, 2) crystalline rocks, 3) argillaceous rocks, 4) carbonate rocks, and 5) volcanic rocks.

Extensive investigation of bedded salt deposits identified potential sites for a waste repository in New Mexico. Exploratory boreholes were drilled at the site in 1974; stratigraphic data indicated that salt horizons existed at about 580, 640, and 820 m (1900, 2100, and 2700 ft) below the surface. In early 1975, this site was selected by the Energy Research and Development Administration (ERDA) for evaluation as a geologic repository site, primarily for disposal of defense program transuranic wastes. Evaluation of this site is continuing under the Department of Energy (DOE).

Currently, the Office of Nuclear Waste Isolation (ONWI) operated by Battelle Memorial Institute for DOE is conducting broadly-based studies on potential repository media related to thermal analysis, rock mechanics, rock-waste interactions, waste migration, risk analysis, etc., as well as repository design and site selection studies. They are currently engaged in investigation of sites for a repository in a salt formation with plans for investigation of other geologic media.

Ongoing studies in the Columbia River Flood Basalt are being conducted by Rockwell Hanford Operations while similar studies of the Nevada Test Site shales and granites are being conducted by the DOE Nevada Operations Office.

REFERENCES FOR SECTION 7.1

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2. R. L. Bradshaw and W. C. McClain, eds, Project Salt Vault: A Demonstration of the Disposal of High-Activity Solidified Wastes in Underground Salt Mines, USAEC Report, ORNL-4555, Oak Ridge National Laboratory, Oak Ridge, TN, 1971.

3. Disposal of Solid Radioactive Wastes in Bedded Salt Deposits, Report by the Committee on Radioactive Waste Management, National Academy of Sciences-National Research Council, Washington, DC, 1970.

7.2 GEOLOGIC CONSIDERATIONS FOR REPOSITORIES

7.2 GEOLOGIC CONSIDERATIONS FOR REPOSITORIES

The objective of placing radioactive waste into a geologic rock unit is to isolate the waste in such a way that radioactive nuclides from the waste cannot migrate to the biosphere in amounts that will endanger the public welfare. This goal may be met in different manners at different sites because of the inherent variability of geological environments. Accordingly, each prospective repository location must be carefully studied and evaluated to identify the unique geologic features of that specific site. The suitability of a site for waste isolation can be determined only after these characteristics are adequately understood.

The general requirements for any site are described in this section followed by a description of the location and characteristics of four candidate media: salt, granite, shale and basalt. However, candidate media are not limited to these four types. Other media such as, for example, carbonate rocks may possess the necessary attributes to qualify for selection as repository sites.

7.2.1 General Geologic Requirements for Repositories

Certain broad requirements must be met before any site can be seriously considered. These criteria are universally applicable to any geological setting or region and to any rock type. General descriptions of these requirements follow:

7.2.1.1 Depth of Isolation Level

The isolation level should be located at a depth below ground surface sufficient to isolate wastes and the encapsulating rock unit from phenomena such as fluvial and wind erosion, near-surface circulating ground water, glacial activity, meteorite impacts, and weathering processes.

The maximum depth of isolation would be governed by the relationship of the lithostatic pressures (the weight of the rock column above the storage facility) to the overall host rock strengths. The lithostatic pressures increase with increasing depth. As depth is increased, these pressures would approach the host-rock crushing strength, and the stability of the repository openings would be endangered. From site to site the lithostatic pressures and host rock strength vary considerably. Accordingly, the maximum depth at which the facilities could be placed is site dependent.

7.2.1.2 Properties and Dimensions of Host Rock

The repository site should be located in a host rock unit of low permeability, porosity, and water content. The low permeability of the host rock formation would minimize movement of groundwater in and near the repository and create the maximum possible isolation from the groundwater flow systems which could transport and release radionuclides to the biosphere. The porosity and water content should be sufficiently low that the water within the host rock would not be affected chemically or thermally by the waste-generated heat on a large enough scale to compromise geological containment and operational safety.

The chemical and radiological properties of the host rock should be such that reaction to the stored waste and containers would not jeopardize the geological containment and operational safety of the repository.

The host rock should have sufficient thickness and lateral extent to provide adequate space to develop and operate the repository, and leave a buffer zone of undisturbed rock on all sides. In addition, the repository rock unit should be of sufficient dimensions to preclude breaching of geological containment by radioactive heat production and possible rock failure or cavity closure after the wastes are emplaced. Minimum thicknesses on the order of 60-75 m (200-250 ft) have been considered during this study; however, greater thicknesses would be more favorable from a rock mechanics and hydrologic standpoint.

7.2.1.3 Hydrogeology

The repository should be located in a region where the groundwater hydraulic gradients are low. This would aid in reducing the rate at which groundwater would leave the repository site. In addition, the repository should be situated far away from any point where the site groundwater flow system discharges to the biosphere or is used by man.

7.2.1.4 Tectonic Stability, Faulting, and Seismicity

A long history of tectonic stability is a prerequisite for a potential repository site. The rate and amount of predictable long-term regional uplift and/or subsidence of bedrock should not pose a threat to the physical integrity of the repository. Faults should be absent from the site or, if present, should not have any characteristics that would adversely affect the operational safety or geological containment. The repository should not be sited in or near an area in which igneous or volcanic activity has occurred during the post-Pliocene (a time period dating from the present to about one and a half to two million years ago). Also, the site should be within a region having a seismic activity level low enough so as not to pose a threat to the safety of the operations or to geological containment. Such a location would be expected to be in a region remote from recorded or historic earthquakes of greater than "moderate" intensity. It should be clearly demonstrated that any structural features found onsite are not associated with potentially active features in adjacent regions.

7.2.1.5 Relationship to Natural Resources

The repository site should be in an area of low resource potential to avoid the risk of future accidental exposure of the repository or the loss of a potentially valuable resource. In addition, a sufficient area surrounding the site should be free from man's activities such as borings, mines, wells, and land development which could endanger the short- and long-term integrity of the waste facilities.

7.2.1.6 Multiple Geologic Barriers

The repository medium and site should provide multibarriers that will enhance isolation of radioactive wastes. The multibarrier concept of isolation provides that the components of a repository site's geologic environment act together to provide isolation. Geologic multi-barrier components are the depth, the disposal medium, media above and below the host rock, the hydrologic regime of the repository medium, the tectonic stability, and the low resource potential.

7.2.2 Potential Geologic Formations

Selection of a disposal medium and repository site depends on earth material properties in an environment that satisfies the general geologic requirements for isolation. The isolation potential of a medium entails the assessment of its interacting physical, chemical, and structural properties. Typical study media having properties that conform to geologic selection requirements include, but are not restricted to, salt deposits (bedded and dome), granite, shale, and basalt.

7.2.2.1 Salt Formations

Figure 7.2.1 shows the major rock salt formations in the United States. Of these, the Michigan and Appalachian basins, the interior province of the Gulf Coast dome region, and the Permian Basin are all considered to be formations of high potential for location of terminal storage facilities. *In the southwest corner of the Permian Basin, the crosshatched section* designated as the Delaware Sub-Basin is the general area of the proposed Waste Isolation Pilot Plant (WIPP facility).

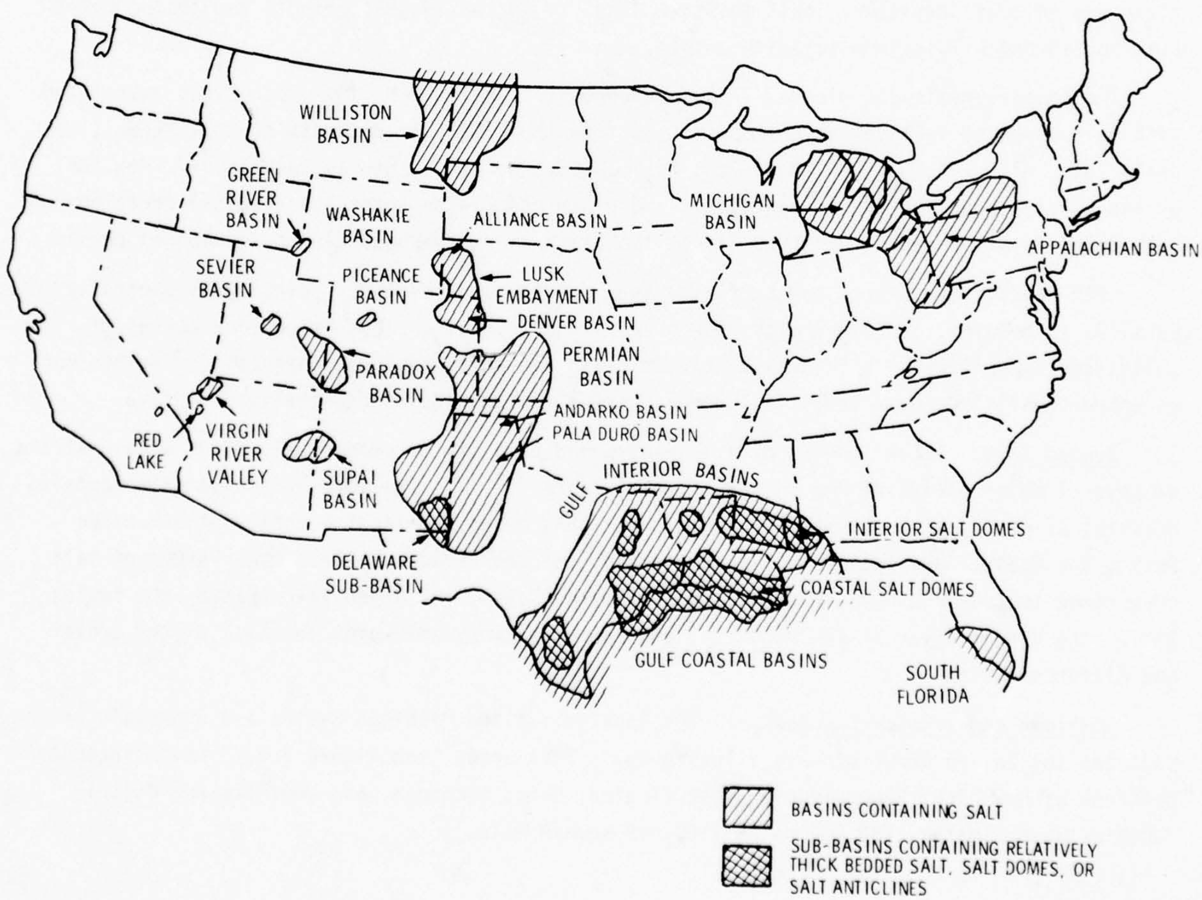

FIGURE 7.2.1. Bedded Salt Deposits and Salt Domes in the Contiguous United States (adapted from Reference 1)

Salt (NaCl) deposits appropriate for disposal media occur in stratiform masses (bedded salts) and salt domes that are formed by precipitation of salt from the evaporation of seawater. Salt precipitation often alternates with the deposition of shale and carbonate minerals, resulting in salt deposits interbedded with other sedimentary rocks. Generally the degree and mineralogical type of interbedding vary greatly. Salt domes are formed by the flow of bedded salts laterally to form masses which then move upward to penetrate overlying strata (diapirism). Salt flow is induced by the low specific gravity of salt plus variations in the lithostatic pressure and differential compaction of overlying sediments. Salt dome deposits are usually of higher purity and are more homogeneous than bedded salts.

The existence of salt beds and formations that are known to be hundreds of millions of years old testifies to their isolation from water and their stability. Salt deposit strength properties are relatively fair to good in the undisturbed state. Salt is basically isotropic with minimal cohesive strength. The result is a highly plastic medium that tends to move (creep) under earth pressures. Creep tends to seal discontinuities but is difficult to stabilize in tunnel openings. Although heat tends to reduce strength, high thermal conductivity of salt is conducive to heat dissipation. A salt deposit may contain moisture in interbed materials and in small cavities. Salt moisture leads to increased heat effects and to the potential for strength loss from solution action.

The major constituent mineral in a salt deposit is halite (NaCl). Rock types associated with salt deposits include anhydrite ($CaSO_4$), limestone ($CaCO_3$), dolomite ($CaCO_3$, $MgCO_3$), and shale (Si_2, Al_2O_3, Fe_2O_3, FeO, MgO, CaO, Na_2O, K_2O). Halite is highly soluble.(2) How ion exchange rate, reaction to radioactivity, and other associated potential chemical reactions of salt deposits and related rock type affect isolation are not adequately understood at present.

Salt deposit structures can be flat-lying, folded, and jointed. Jointing is generally parallel to bedding. Included within beds are large crystal masses, large rock masses of solidified impurities with lateral continuity, and lateral lithologic changes.(3) Joints can be anhydrite-filled, near vertical, unopen, moderately spaced, and generally extensive.

Bedded Salt. Thick, geographically widespread deposits of bedded salt occur within strata of several major basins of the United States (Figure 7.2.1). The more extensive salt deposits occuring at depths of interest lie within the Michigan and Appalachian Basins, the Permian Basin, the Coastal and Interior Gulf dome areas, and the Paradox Basin. Other sites of salt occurrence include southwest Florida, the Williston Basin, the Green River Basin, the Sevier Basin, the Virgin River Valley deposits, and the Lusk Embayment which consists of the Denver and Alliance Basins.

Michigan and Appalachian Basins. The Appalachian and Michigan Basins are underlain by the salt bearing Salina Group of Late Silurian age. This area, (see Figure 7.2.1) which includes portions of New York, Pennsylvania, West Virgina, Ohio, Michigan, and southwestern Ontario contains approximately 260,000 sq km (100,000 square miles).

The Salina Group consists of, from oldest to youngest, the Vernon red and green shales, the Salina salt (interbedded series of dolomite, shale, salt, and anydrite) and the Camillus gray shale, dolomite, and gypsum.[4] The Pittsford shale, which locally underlies the Vernon shale, and the Bertie limestone, which rests upon the Camillus shale, have been included within the Salina Group by some investigators.[5] Generally, a salt series consists of individual beds of salt, separated by interbeds of shale, dolomite or limestone, gypsum and/or anhydrite. In New York, the formation contains as many as six beds of salt. In Michigan, the Salina salt series is divisible into seven recognizable units and may contain eight salt beds.[5] The Salina Group in northeastern Ohio contains four major salt zones (total of 8 salt beds).[5,6]

The depth to the top of the salt series varies from near ground surface in a line extending east-west across much of western New York, to more than 1370 m (4500 ft) below sea level south of Binghamtom, New York. The salt series is thicker than 400 m (1300 ft) south of Lake Seneca, but individual salt beds vary in thickness up to 170 m (550 ft).[4,7] In New York the salt is generally overlain by a sequence of limestone, dolomite or shale and underlain by shale.

The Salina Group in Pennsylvania (Figure 7.2.1) occurs at depths greater than 1200 m (4000 ft) below ground surface.[8] In this part of the basin the Salina Group consists of salt, dolomite, dolomitic limestone, dolomitic shale, and anhydrite. The number of individual salt beds varies from eight in north-central to two in southwestern Pennsylvania.[5] Individual salt beds may be as thick as 60 m (200 ft).

In Ohio the Salina Group crops out along the east limb of the Findlay Arch and dips generally toward the southeast. Consequently, the depth is variable but generally is greater than 1800 m (6,000 ft). Salt beds are most numerous in north-eastern Ohio where eight salt beds have been recognized within the formation. In this area the Salina Group consists of salt, dolomite, dolomitic limestone and subordinate shale.[5] The thickest individual salt bed is approximately 15 m (50 ft) thick and the fourth salt zone is less than 5 m (18 ft) thick.[6]

The stratigraphy of the Michigan Basin is generally similar to that of the Appalachian Basin.

The Michigan Basin contains bedded salt deposits within the Silurian Salina Group and the Devonian Lucas Formation. The Salina Group underlies most of Michigan's southern peninsula and connects with the Appalachian Basin via Ohio and southwestern Ontario (see Figure 7.2.1). The Salina salt, however, does not underlie the entire Michigan Basin. Whereas the Salina Group reaches a thickness of almost 900 m (3,000 ft) at a depth greater than 2000 m (6,500 ft) in central Michigan, the salt series and individual salt beds become thinner, pinch out, and slopes up, toward the rim of the basin.[5,9]

Bedded salt in the Michigan Basin also occurs within the Devonian Lucas formation of the Detroit River Group. Only the northern part of the southern peninsula contains Devonian salt. The Lucas salt series consists of salt interbedded with anhydrite and shale. This sequence, however, contains relatively thin salt beds which are often measured in tens of feet or occur at depths greater than 900 m (3000 ft).[5,10,11]

The Permian Basin (Including Delaware Sub-Basin and Supai Basin). The Permian Basin (Figure 7.2.1) consists of a series of sub-basins separated by either domes or arches. Approximately 310,000 sq km (12,000 square miles) are underlain by Permian age salt.[5] The Delaware sub-basin is a major downwarp within the Permian Basin. The oldest salt is the Lower Permian (Leonardian Series) Hutchinson Salt, which occurs in the Anadarko Basin of Oklahoma and Kansas. Younger salts occur in the Permian Basin with the Youngest (Upper Permian Ochoan Series) occurring in the Palo Duro and Delaware sub-basins. Outcropping salt may exhibit evidence for solution that extends into the subsurface.[12,13]

In general, some individual salt beds are thicker than 75 m (250 ft) in parts of the Anadarko, Midland and Delaware sub-basins.[12,14] The Hutchinson salt is thicker than 90 m (300 ft) in parts of Kansas.[15,16]

Salt in the Supai or Holbrook Basin of Arizona and New Mexico (Figure 7.2.1) consists of an upper and lower salt in the upper Supai formation, and a salt within the lower member of the Supai formation.[17,5] Individual salt beds do not exceed 50 m (160 ft) in thickness; however, the upper Supai contains an aggregate thickness of up to 150 m (485 ft) of evaporite (mainly halite).[18] Available, published subsurface data on the upper Supai salt indicates widespread lateral occurrence of halite, but at depths less than 550 m (1800 ft) below ground surface. In addition, some salt appears to have been dissolved in the shallow subsurface.[18]

A comparison between stratigraphic sections of the Permian and the Michigan and Appalachian Basins indicates that shale often overlies salt in the Permian Basin whereas dolomite or shale overlie salt in the Michigan and Appalachian Basins.

Southwest Florida. Bedded salt in southwest Florida (Figure 7.2.1) occurs at depths greater than 3,300 m (11,000 ft). Total thickness appears to be less than 9 m (30 ft).

Williston Basin. The Williston Basin is located in part of southern Canada, North and South Dakota, and eastern Montana (Figure 7.2.1). Eleven salt beds have been located within the sub-surface. The stratigraphically highest and youngest salt is the Jurassic Dunham salt. The oldest and deepest salt is the Devonian Prairie formation. Depth to salt, in all cases, is greater than 600 m (2,000 ft) below the surface.[19-23]

Green River Basin. Eocene salt is associated with trona deposits of the Green River Basin in southwestern Wyoming (Figure 7.2.1). In the evaporite-bearing portion of the basin, the Green River formation has been divided into three members. The middle member, the Wilkins Peak, contains salt in its lower portion. This salt varies in thickness and lateral extent.[24]

Evaporites are associated with the Parachute Creek (upper) member of the Green River formation in the Piceance Basin of northwestern Colorado. These salts either occur at depths greater than 600 m (2,000 ft) below ground surface or are immediately overlain by a zone leached by groundwater. In addition, most oil shale occurs within the Parachute Creek member.[25]

Sevier Basin (Valley). The Sevier Valley is located in central Utah (Figure 7.2.1). Bedded salt occurs within the youngest unit of Jurassic Arapien Shale.[26] This unit has salt beds which probably are over 60 m (200 ft) thick and contain relatively high proportions of red clay and silt.[26,28] Due to structural complexities, such as folding and faulting, the Arapien salt may occur at depths of 550 m (1,800 ft). However, published subsurface data were not available to substantiate this.

Virgin River Valley Salt. The Virgin River Valley Salt (Figure 7.2.1) locally occurs within a 490 m (1,600 ft) section of the Pliocene (?) age lower Muddy Creek formation (dominantly siltstone, claystone and sandstone).(28) These evaporites were deposited in a playa lake environment. The Virgin River Valley salt is exposed along or near the shore of the Overton arm of Lake Mead in southern Nevada and was also exposed along the Virgin River prior to construction of Hoover Dam and the formation of Lake Mead.(28,29) These salt deposits either outcrop and are exposed to groundwater solution or occur within several salt anticlines which are commonly cross-cut by normal faulting.(28) The contact between salt and overlying gypsum steepens to a dip of 60-70 degrees on the flanks of one salt dome.

A salt deposit, probably equivalent to the Virgin River Valley Salt occurs in the Red Lake paleobasin in Hualpai Valley, Mohave County, Arizona. The El Paso #1 well at Red Lake encountered salt at a depth of 550 m (1,800 ft) below ground surface and continued in salt to a depth of 1,800 m (6,000 ft).(30) This deposit appears to be a diapir implaced in post-Miocene time.(31) However, specific contact relationships have not been clearly established.

Lusk Embayment (Denver and Alliance Basins). The Lusk Embayment, as shown on Figure 7.2.1, is a combination of the Alliance Basin in Nebraska and part of the Denver Basin in Colorado and southwest Nebraska. Salt in the Alliance Basin is found in the Permian Goose Egg Group and ranges in depth from 970 to 1,800 m (3,200 to 5,900 ft) below ground surface.(32)

The Permian section within the eastern Denver Basin is relatively thin. Evaporites are present but salt is absent or negligible. If halite is present it occurs at depths greater than 880 m (2,900 ft).(32, 33, 34)

Salt Domes and Salt Anticlines. Salt domes and salt anticlines are structures with a core of massive salt. The salt has flowed laterally and risen upward from its original stratiform position. In a dome the salt has broken through overlying rock layers; in an anticline, it has not. The triggering mechanism for salt flow is thought to be the density contrast between salt and overlying sediments and differential compaction of overlying sediments. The less dense salt flows upward in response to the increasing confining pressure.(5) The salt must be deeper than a critical depth and thicker than a critical thickness before flow can begin.

The major areas of salt dome occurrence in the United States are the Gulf Coast and Gulf Interior Basins (Figure 7.2.1). The Paradox Basin in southwest Colorado and southeast Utah is the major area of salt anticlines in the U.S.

Gulf Coast and Gulf Interior Domes. The salt which forms domes in the Gulf region may have originated in either the Jurassic or Permian Louann salt or the stratigraphically higher Jurassic Buckner formation.(35) These bedded salts occur at depths greater than 900 m (3,000 ft) in the interior basin and at depths greater than 3,000 m (10,000 ft) toward the Gulf Coast.(35-37) Sediments above salt domes are deflected upward indicating an upward movement of salt. Normal faults above or on the flank of several salt domes provide further evidence for upward salt movement.

The presence of stratigraphic units that are thin above the salt domes and thicker away from the dome suggest that the rise of the dome has been, in part, contemporaneous with the deposition of these sediments.

Paradox Basin. In the Paradox Basin, the Paradox salt member occurs within the Pennsylvanian Hermosa Group. This bedded salt usually occurs at depths greater than 1,500 m (5,000 ft), but does occur at depths less than 600 m (2,000 ft) below ground surface on the flanks or over the core of several salt anticlines.[32,38,39] The thickness of the salt layer varies from nearly zero up to about 2,500 m (8,000 ft) depending on whether the salt has flowed from or to the specific area.

7.2.2.2 Granite Formations

Large granitic rock bodies occur within the seven areas listed below and shown in Figure 7.2.2.

- Northern Appalachians - New England (Maritime)
- Central and Southern Appalachians
- The Precambrian Shield of the Great Lakes area
- Northern and Southern Rocky Mountains
- The Sierra Nevadas of California
- Arizona (Basin and Range) Granites
- Idaho and Boulder Batholiths

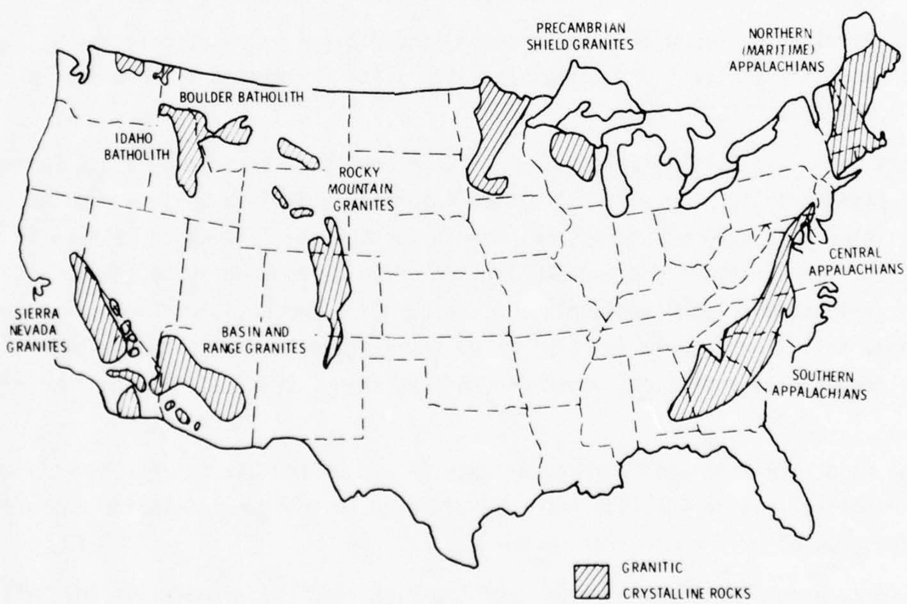

FIGURE 7.2.2. Granitic Rock in the Contiguous United States (adapted from Reference 1)

Granite is an intrusive igneous rock having an equigranular, medium-to coarse crystalline texture. It is light colored, composed of feldspar, quartz and, typically, hornblende and biotite. Granites are generally homogeneous in composition, with variations primarily in accessory minerals and secondary rock features. Granites are found as plutons, which are bodies of igneous rock that

have formed beneath the earth's surface by consolidation from magma. Typical granite plutons include batholiths and smaller-scale stocks with the distinct characteristic that they are relatively bottomless and enlarge with depth.[2,40]

Igneous rocks may have similar physical characteristics but range in chemical and mineralogical composition from granite to closely related rocks such as granodiorite. In many respects other closely related igneous rocks are similar to granite, but, because they vary significantly in major element, trace element and mineralogic composition, they are not considered to have the same disposal media properties as granite.

Because granite is formed beneath the earth's surface its texture is a dense matrix of equigranular coarse grains. The porosity is low, with little to no natural moisture content. Intergranular permeability is extremely low. Also strength is considered to be very high. Most component minerals are hard, resulting in high durability. Granites are generally very rigid, with little ability to deform under earth stress. Granites are basically unaltered by heat because of the high temperatures of formation. However, thermal expansion of particular minerals may be sufficient to cause ruptures of rock and surface heave.

Granite is mostly composed of silica and mica. Typical chemical composition of granite is contained in Table 7.2.1.[2]

TABLE 7.2.1. Average Chemical Composition for Nonsalt Disposal Media (reported as oxides)

Compound	% of Total Granite	Shale	Basalt
SiO_2	77.0	55.0	49.1
Al_2O_3	12.0	21.0	15.7
Fe_2O_3	0.8	5.0	5.4
FeO	0.9	1.5	6.4
CaO	0.8	1.6	9.0
Na_2O	3.2	0.8	3.1
K_2O	4.9	3.2	1.5
MgO	---	2.3	6.2
H_2O	0.3	8.1	1.6
X_yO	0.1	1.9	2.0

Mineral components of granite are almost inactive chemically under ambient temperature and pressure conditions. However, the chemical reactiveness of granite has not yet been fully investigated over the range of expected repository conditions.

Granites have no bedding because of their intrusive igneous mode of formation, but may be layer-like. Joints tend to be blocky on a large scale, and their orientations are generally vertical and intersect at right angles. Joints often have little mineralization and range from sealed to partially opened and extensive.

7.2.2.3 Shale Formations

Shale (Figure 7.2.3) is the product of the lithification (compaction and cementation) of mud. Mud is predominantly composed of clay size particles (1/256 mm dia) and/or silt size particles (1/256 to 1/16 mm dia). The predominant constituents are clay minerals (hydrous aluminum silicates), and substantial amounts of mica, quartz, pyrite, and calcite (Table 7.2.1).[2,40] Mineral grains may either be poorly compacted in a soil-like manner or cemented like rock. Shales are in general stratified or laminated, and fissile, although some may show little layering and break into small angular blocks, as with mudstones. Shales are often interbedded with other sediments such as carbonates and sands.

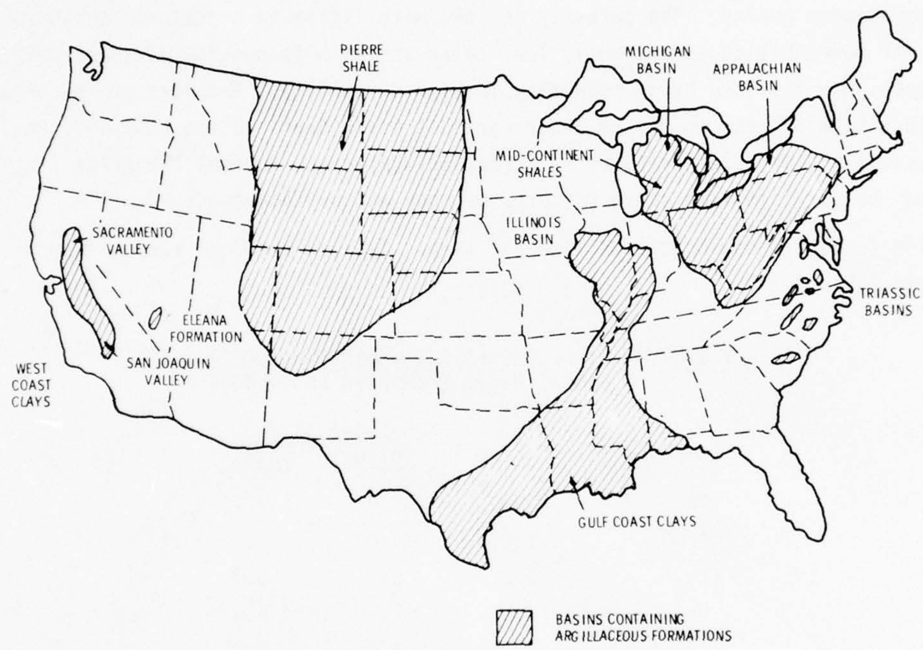

FIGURE 7.2.3. Shale Formations in the Contiguous United States (adapted from Reference 1)

Shales are relatively weak, partly because of the soft mineral components and weak cementation between grains. The general texture is fine-grained with a fissile structure, causing the shale to split into flat, shell-like fragments in parallel bedding. The fine-grained clay minerals account for a very high natural moisture content and porosity. Voids are not interconnected so permeability is low. Many shales have the ability to accommodate large deformations with a potential for plastic flow.

Clay minerals are known to have a high ion-exchange potential. Wetting and drying of shale will weaken the rock and may cause slaking. Shale's chemical reactivity is not yet adequately understood for waste isolation purposes.

Shales may have discontinuities consisting of bedding, joints and fracture planes that are often filled with calcite, but they also may be unfilled.

Mid-Continent Shales. Relatively thick marine shales in the mid-continent region (Figure 7.2.3) vary in age from Ordovician to Mississippian. In central Ohio the Upper Ordovician Cincinnatian series contains shales interbedded with argillaceous limestones. The shale becomes less calcareous toward the southeast. These shales are generally thicker than 60 m (200 ft) and may be encountered at depths ranging from 300 to 450 m (1000 to 1500 ft) on the flanks of the Cincinnati-Findlay Arch.[41,42] These rocks are generally underlain by Cambro-Ordovician carbonates (Trenton and Black River limestones) and are overlain by argillaceous carbonates.[42]

The tops of thick Devonian marine shales occur at depths between 300 and 450 m (1000 and 1500 ft) in east-central Ohio, central Indiana and Michigan.[9,42] This carbonaceous shale (Ohio shale in Ohio, Antrim shale in Michigan) is over 340 m (1100 ft) thick in central Ohio and becomes thicker toward the southeast.[42,43]

In western Michigan the black carbonaceous Antrim shale interfingers with, and is replaced by, the greenish gray to black Ellsworth shale. In parts of western Michigan the Antrim and Ellsworth shales are overlain by the Sunbury and Coldwater shales. This shale sequence is thicker than 240 m (800 ft) in west-central Michigan.[9,44] The Coldwater is dominantly a silty and calcareous shale interbedded with small amounts of siltstone, sandstone, limestone, and dolomite.[44]

The Devonian shale sequence is generally underlain by limestone (with and without chert) and overlain by argillaceous limestone, shale with minor siltstone, sandstone or shale.[43] The Ellsworth shale is both overlain and underlain by shale. The Mississippian Coldwater shale is overlain by either the Marshall sandstone or glacial deposits.[9,44]

Triassic Basin (Virginia, North Carolina, South Carolina). The Triassic Basins of Virginia, North Carolina and South Carolina (Figure 7.2.3) contain continental deposits. In general, silt-shales, mud-shales and siltstones tend to be thin and discontinuous. However, relatively thick, possibly laterally continuous shales may occur at depths greater than 600 m (2000 ft) in the Richmond Basin of Virginia and Deep River Coal Field in North Carolina. Cumnock formation shales occur at depths between 300 and 500 m (1000 and 1700 ft) in the Deep River Coal Field and can reach thicknesses of 120 m (400 ft). However, they are associated with the Cumnock Coal.[45]

Pierre Shale. The Pierre shale and its lateral equivalents underlie an area of 960,000 sq km (370,000 square miles) as shown in Figure 7.2.3. The Pierre consists of thick sequences of claystone, shale, bentonitic mudstone and many bentonite beds. This shale unit ranges in thickness from 150 m (500 ft) in the eastern Dakotas to more than 1500 m (5000 ft) in southeastern Wyoming and central Colorado.[44] In several areas of Montana, Wyoming, North Dakota and South Dakota, the Pierre consists of relatively thick sequences of impermeable marine shale greater than 300 m (1000 ft) thick. In other parts of these four states, the Pierre contains beds consisting of sandstone, organic-rich shale, silty shale, calcareous shale, marlstone and/or impure chalk.[32]

Eleana Formation. The Eleana formation of Mississippian age occurs at the Nevada Test Site in Nye County, Nevada (Figure 7.2.3). This formation consists of argillite, siliceous siltstone and very fine grained quartzite, quartzite and conglomerate and limestone.[46,47] Facies changes occur rapidly within a short distance making the Eleana formation lithologically inhomogeneous. This Eleana is underlain by Devonian dolomite, limestone or sandy limestone and is overlain (where not exposed) by Tertiary tuff, pediment gravel or Pennsylvanian to Permian limestone.

7.2.2.4 Basalt Formations

Basalt is a black to medium gray, extrusive volcanic mafic (high in magnesium rock silicates) rock. Its major mineral components are calcic plagioclase (usually as phenocrysts) olivine and accessory minerals of magnetite, chlorite, sericite, and hematite.[2] The texture of a basalt may be either glassy or granular. Generally basalt flows have a large areal extent.

Basalt is a very dense, high-strength material. Consequently, porosity and permeability are favorably low, with a negligible moisture content. Basalts remain relatively strong under elevated temperatures but may expand. An average chemical composition of basalt is included in Table 7.2.1. The reactive nature of basalt has not yet been adequately investigated for isolation purposes.

Joints are generally platy or columnar. They may be filled with various secondary minerals, alteration, or weathering products of basalt. Joints may be unopened or opened with wide spacing ($\sim$0.3-1.8 m) and be smooth to rough. Joints in basalt may be extensive.

Flood basalts are thick and extensive in certain parts of the United States (Figure 7.2.4). The thickness is the result of many superposed individual layers, each representing an eruptive event. Interludes between eruptions caused erosion of flows and resulted in the deposition of sediment.

Flood basalts occur extensively in the Pacific Northwest; mainly southeastern Washington, northeastern Oregon, and adjacent Idaho. The basalt in these areas averages 100 m (3200 ft) thick over approximately 200,000 sq km (80,000 square miles). The maximum known thickness is 3,250 m (10,600 ft). Most basalt accumulated as superposed flows from Miocene to early Pliocene time - nearly 8 million years ago. The source is believed to be a swarm of dikes that are located along the eastern margin of the region.

7.2.3 Generic Site Descriptions

Each of the four geologic media described in Section 7.2.2 have unique characteristics that result in different repository design, construction, operation and decommissioning requirements. To develop the conceptual repository designs described in this report, generic site descriptions were derived for the four media.

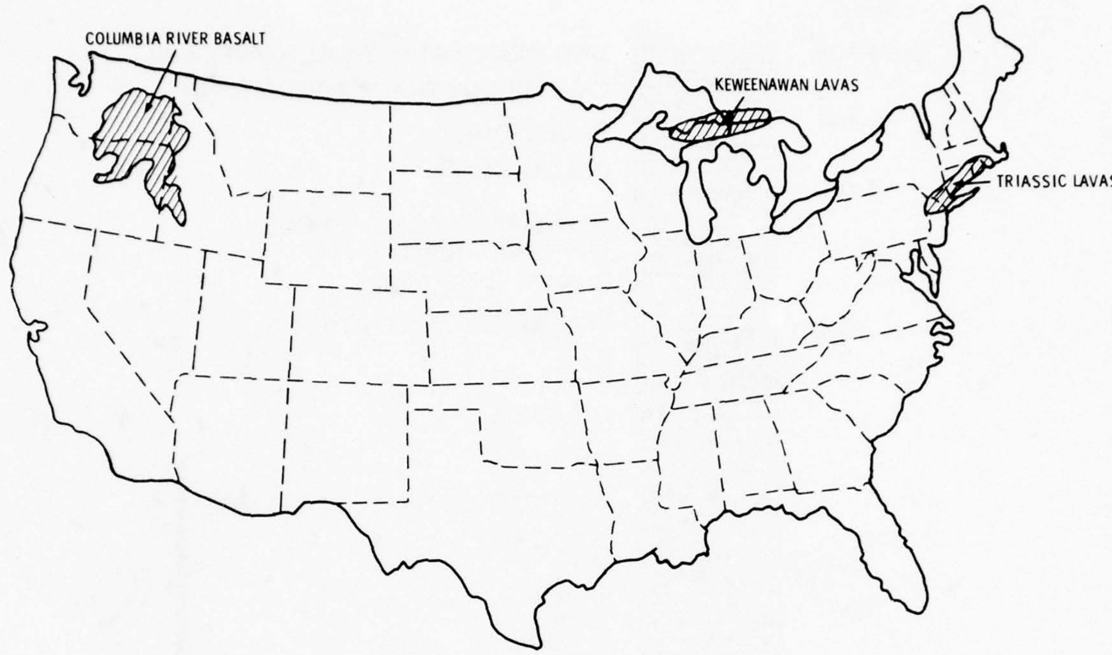

FIGURE 7.2.4. Potential Repository Basalts in the United States (adapted from Reference 1)

7.2.3.1 Bedded Salt Site

To satisfy the geologic guidelines related to the stability of the repository openings and long-term containment (Section 7.2.1), a bed of salt at least 76 m (260 ft) thick above the 600 m (2000 ft) depth will be needed to house the repository. Detailed examination and comparison of the various basins and stratigraphies described in Section 7.2.2 indicated that these conditions can be found in several areas in the United States. A generic section (which excludes salt domes and anticlines discussed in Section 7.2.2) representing the general stratigraphy which can be expected to be encountered at a bedded salt repository is shown in Figure 7.2.5.

The generic section was derived as follows. First, using the depth and thickness requirements, certain basins were eliminated from consideration. Next, representative sections for the remaining areas were developed by combining several sections from published literature into a composite section for each area. Finally, the generic section was obtained by integrating these composite sections. The sequence of formations in the generic section was found to occur in most of the basins examined. Variation in the actual versus generic sections represents different stages in the restriction of a seaway and the resulting concentration of soluble salts and depositional environments. The generic stratigraphic section is considered a typical evaporite sequence.

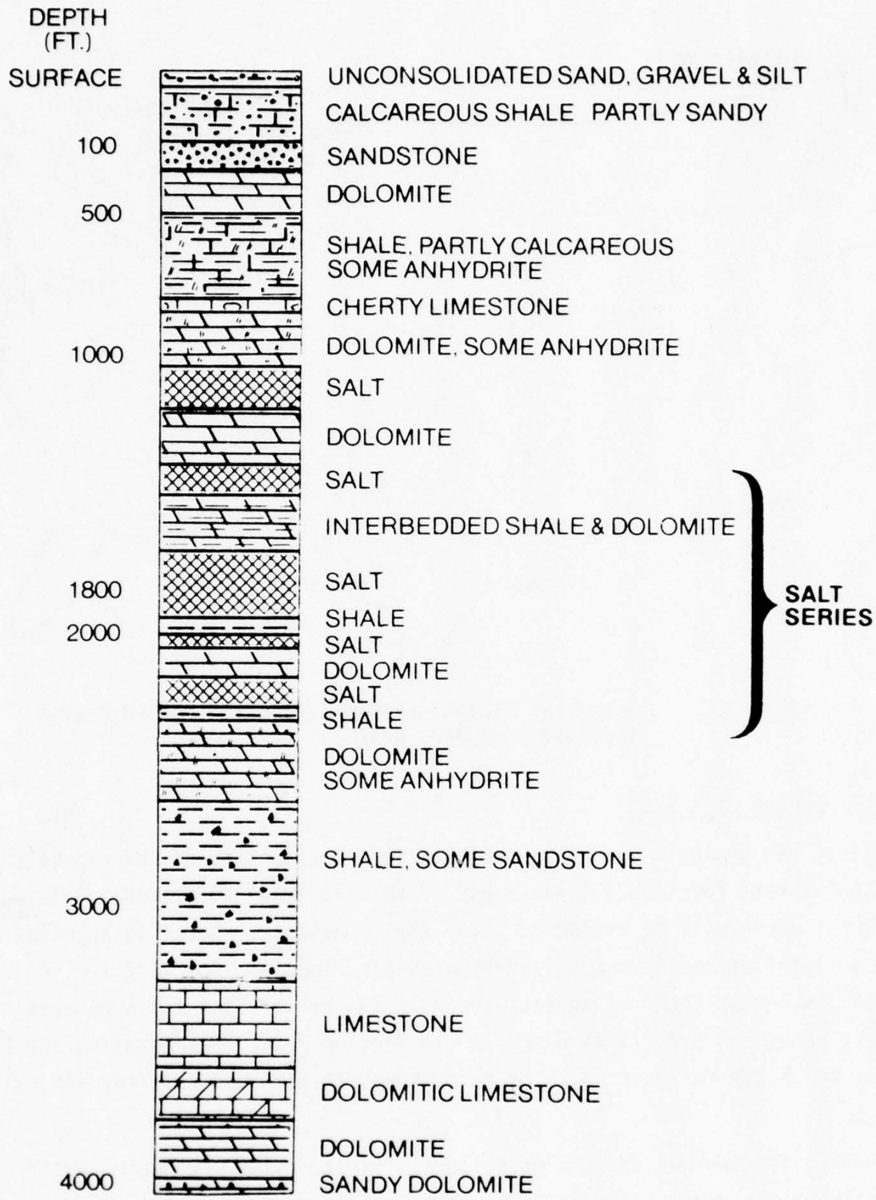

FIGURE 7.2.5. Generic Stratigraphic Section--Salt

Hydrogeology

The generic bedded salt section presumably is located in one of the large undeformed geologic sedimentary basins. The strata are nearly flat-lying and the regional dip is less than one degree. The recharge areas are topographically not much higher than the discharge areas, so the regional groundwater gradient is low.

The generic section contains a sequence of clastic and/or evaporative rock formations with varying permeabilities and porosities. These water-bearing formations, which overlie and underlie the repository host rock, contain and transmit various quantities of groundwater, depending upon the regional gradient and certain hydrogeologic parameters such as porosity, permeability, and formation thickness. The values for these parameters will vary considerably from site to site. A chart listing the ranges of values obtained from a literature search and past experience is shown on Table 7.2.2.

TABLE 7.2.2. Hydrologic Parameters for Generic Salt

Rock Type	Generic Hydraulic Conductivity (cm/sec)	Hydraulic Conductivity Range	Porosity (%)
Terrace deposits glacial till	6.8×10^{-6}	$1 \times 10^{-7} - 1 \times 10^{-3}$	20
Calcareous shale partly sandy	1.2×10^{-5}	$1 \times 10^{-10} - 1 \times 10^{-1}$	13
Sandstone and dolomite	8.1×10^{-5}	$1 \times 10^{-10} - 1 \times 10^{-2}$	20
Shale partly calcareous some anyhdrite	1.2×10^{-5}	$1 \times 10^{-10} - 1 \times 10^{-3}$	13
Cherty limestone dolomite, some anhydrite	5.8×10^{-5}	$5 \times 10^{-8} - 1 \times 10^{-3}$	20
Dolomite	4.6×10^{-5}	$5 \times 10^{-8} - 1 \times 10^{-2}$	30
Salt	Nil	$2 \times 10^{-21} - 1 \times 10^{-8}$	0.5
Interbedded dolomite and shale	6.8×10^{-6}	$1 \times 10^{-10} - 1 \times 10^{-3}$	20
Shale	7.1×10^{-7}	$1 \times 10^{-10} - 1 \times 10^{-3}$	16

In-situ Rock Stresses

Because of the rheological character of salt, a lithostatic stress field (equal in all directions) equivalent to the weight of rock above the salt would normally be anticipated. The reason is that, provided sufficient time has elapsed after any tectonic stress has been applied, the salt will creep and readjust until lithostatic equilibrium is reached. Accordingly, the horizontal and vertical in situ stresses at the repository level of 580 m (1800 ft) will be about 1700 to 1900 psi.

Rock Properties

It is important to distinguish the difference between the properties of a small intact piece of rock (dimensions typically in cm) and the properties of the rock mass (dimensions typically in hundreds of meters) when considering rock properties. Intact properties are usually determined in the laboratory whereas rock mass properties are evaluated in large scale field tests or, more generally, are estimated in some manner by using the intact properties and the rock mass structure as a basis.

A literature search was carried out[3] to find out the range of property values for intact salt. The type of properties reviewed included index properties, stress-strain relationships and strength and thermal properties. Means were calculated for each of the properties and maximum and minimum values were determined.[1]

Although bedded salt formations may be several thousand feet in overall thickness, they usually consist of irregular sequences of clear to white halite with thin laminae of shales, siltstones, clay, gypsum, and anhydrite. The salt layers range from a few inches to 30 feet in thickness. The significant features in the salt are layering of clear and cloudy salt and seasonal layering of shale and anhydrite with occasional thicker beds of shales, anhydrite, and other materials.

Basin salts are characterized by gentle dips. However, they can be folded or faulted. Jointing or fracturing is often due to partings along impurity layers. Otherwise, jointing is rare, because salt is a relatively plastic material and joints within the halite tend to self-seal. For this reason, salt beds are generally impervious, because there are few vertical joints through which the water can penetrate.

A summary of the structures found in a typical salt formation is contained in Reference 1.

Because of the relative lack of structure in salt when compared to other rock types, the rock mass properties do not vary markedly from the intact properties. However, some variations do occur. Based on the methods described in Reference 3, rock mass properties for the generic host salt rock have been estimated and are listed in Table 7.2.3. Additional details of salt's intact and rock mass properties are provided in References 1 and 3.

Environmental Effects. The properties of the salt will be affected by certain environmental factors including time, temperatue, pressure, moisture or water, and irradiation.

Time Effects. A notable characteristic of salt is its creep behavior with time. Creep is defined as viscous flow under constant stress.

Several stages of creep have been delineated. These stages depend upon the time span of the observations as well as the magnitude of the applied stress as shown in Figure 7.2.6. For instance, when the material is initially stressed, there is an immediate elastic strain followed by primary creep in which the strain-time curve is concave downwards. This is followed by secondary or steady state creep, in which the curve has an approximate constant slope. Finally, the tertiary creep corresponds to accelerating strain and leads rapidly to failure.

In general, the primary and tertiary stages are very short, usually less than one day, while the secondary stage is comparatively long and depends upon the magnitude of the applied load.

From experiments carried out in the laboratory, empirical equations can be developed to describe the creep behavior of the salt. These equations are complex and no agreement has been reached as to which is the best one. The important point, however, is that salt does creep and a repository cannot be rationally designed unless the creep behavior under the appropriate conditions of pressure and temperature is properly understood.

TABLE 7.2.3. Estimated Rock Mass Properties for Generic Salt

Type of Property	Parameter	Value
Index	Unit Weight	2128 kg/m^3 (133 lb/ft^3)
	Natural moisture content	0.08%
	Porosity (rock mass)	0.5%
Stress-strain	Young's modulus	1.12×10^5 kg/cm^2 (1.6×10^6 lb/in^2)
	Poisson's ratio	0.35
Strength	Unconfined Compressive Strength	225 kg/cm^2 (3200 lb/in^2)
	Tensile Strength	3.5 kg/cm^2 (50 lb/in^2)
Thermal	Coefficient of Linear Thermal Expansion	40.0×10^{-6} °C^{-1} (22.2×10^{-6} °F^{-1})
	Heat Capacity	
	Temperature, °C	Heat Capacity, W•sec/kg•°C*
	0 (32°F)	840 (0.20 BTU/lb-°F)
	100 (212)	880 (0.21)
	200 (392)	920 (0.22)
	Thermal Conductivity	
	Temperature, °C	Conductivity, W/m °C
	0 (32°F)	6.11 (3.53 BTU/hr-ft °F)
	50 (122)	5.02 (2.90)
	100 (212)	4.20 (2.43)
	150 (302)	3.60 (2.08)
	200 (392)	3.11 (1.80)
	300 (572)	2.49 (1.44)
	400 (752)	2.08 (1.20)

*1 W•sec/kg•°C = 1 J/kg•°C

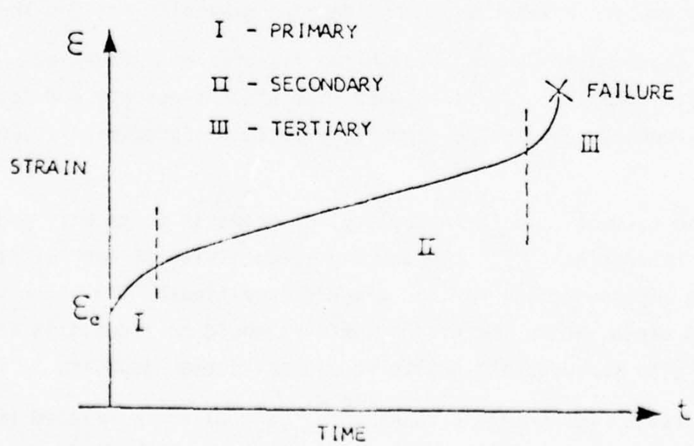

FIGURE 7.2.6. Generalized Creep Curve for Salt Under Conditions of Constant Stress

Temperature. The physical behavior of salt is drastically affected by temperature. For instance, increased temperature increases the rate of creep and decreases the value of Young's modulus. Experiments carried out by Lomenick and Bradshaw,[48] demonstrated that for a rise in temperature from 22.5°C to 100°C the strain increased by a factor of seven.

The drop in compressive strength with increase of temperature is not as marked but is reduced by approximately 10% between 22.5°C and 200°C.

The thermal conductivity decreases and heat capacity increases with increasing temperature.

Pressure. The effects of pressure on the properties of salt have not been extensively investigated or reported. However, it is expected that for an increase in deviatoric stress the creep rate of the salt will increase.

Water/Moisture. Moisture is often trapped internally in vesicles and along partings. Most of the moisture is brine. The trapped brine can be released with considerable energy when heated and can fracture the rock. In laboratory tests, salt from mines in Kansas fractured at temperatures of 260°C to 320°C.[48]

Irradiation. The effect of irradiation on bedded salt was studied by Bradshaw et al.[49,50] It was concluded that radiation exposure doses of 5×10^8 rems produce some changes in structural properties, but the effect on mine stability should be negligible.

7.2.3.2 Granite Site

The representative stratigraphy selected for the generic waste repository in granite consists of continuous granitic rock with a thin cover of weathered material. Joint systems are expected to be present in the granitic rock mass. The orientation and frequency of joints and the presence or absence of shear zones are a function of the tectonic history and cooling history of the specific region within which the pluton is located. The generic jointing conditions are described under Rock Properties below.

Hydrogeology. The generic granite section is located in an area where the topographic relief is very low over great distances. This low topographic relief results in a low groundwater gradient as the groundwater is unconfined and generally follows the topography.[50]

The flow of groundwater occurs through the fissures and joints and/or singular disconformities in the granite body.[50] It is assumed that these fractures and joints are open to the atmosphere and extend as deep as the repository located approximately 600 m (2000 ft) below land surface.

The hydraulic conductivity (permeability) of granitic rocks will generally be very low. As discussed in the references[50,51] a hydraulic conductivity of approximately 10^{-9} to 10^{-8} cm/sec was assumed to be representative for the generic conditions. The lower values are representative of the granite at depth, where the discontinuities would be reduced in size and frequency because of the higher in situ stresses and unaltered mineralization deposits in fractures.

Values of porosity for granitic rocks[50,51] are directly related to the frequency and size of fractures and joints. Unless the rock is unusually weathered and altered by geologic or tectonic activity, it is unlikely that the porosity would exceed 1%. As discussed in the technical support document,[51] a value of 0.005 (1/2%) was chosen for the porosity of the generic granitic host rock.

In-situ Rock Stresses. The in-situ stresses in granitic rocks are dependent upon depth and previous stress history, particularly tectonic activity. The vertical stress is usually at least equal to the overburden pressure. For the generic conditions it has been assumed that the vertical stresses will be equal to the overburden. The horizontal stress is usually expressed as a fraction of the vertical stress; this fraction lies in the range of 0.5 to 2.0 and is generally considerably greater than 1.0 for granite. Accordingly, a value of 1.5 can be considered as representative of the generic site conditions.(52)

Rock Properties. In order to obtain an idea of the properties of intact granite, a literature search(51) was carried out similar to that for the salt. The results are summarized in Reference 1.

The unit weight is fairly constant at about 165 lbs/ft^3 and the natural moisture content and porosity are very low. In these competent rocks Young's Modulus is high, as are the compressive and tensile strengths.

Granites comprise some of the more competent rocks occurring in the earth's crust. However, shear zones are frequently encountered in the rock mass and these have caused serious support and groundwater flow problems in underground chambers and tunnels.

Granites sometimes exhibit pronounced anisotropy because of the presence of a number of small cracks. While this anisotropy affects the properties of the intact material, granite is less likely to be anisotropic at the scale of the rock mass.(51) Although joint orientations may appear to be randomly oriented, definite trends are sometimes apparent. At a depth of 600 m (2000 ft) and below the ground surface, joints and fractures tend to be closed.

A description of structures in the generic granite is contained in Reference 1. It is assumed that these types of structures are present in the generic granite medium.

Based on the intact properties, the assumed structure, and a detailed review of the rock mass properties of several granites, the values outlined in Table 7.2.4 are selected as properties to be used in the conceptual design studies.

Environmental Effects. The properties of granite will be affected by environmental factors.

Time. Very few tests of long duration have been carried out on granites, and the time-dependent behavior can only be inferred from triaxial compression tests. A creep law for granite has been determined; however, the effects of creep in granite are very small when compared to salt or shale.

Temperature. The effect of increased temperature on granites commonly produces lower yield stresses, reduced moduli values and a reduction in Poisson's ratio. For instance, a rise in temperature of 200°C reduces Young's modulus to almost one-half and Poisson's ratio to one-fifth of its value at room temperature.(51)

Pressure. The effect of confining pressure on granite is to increase the ultimate strength and the Young's modulus value. Poisson's ratio tends to decrease slightly with an increase in pressure, although a dry granite would probably show a pronounced initial increase prior to a decrease.

TABLE 7.2.4. Estimated Rock Mass Properties for Generic Granite

Type of Property	Parameter	Value
Index	Unit Weight	2640 kg/m^3 (165 lb/ft^3)
	Natural moisture content	--
	Porosity (effective)	0.4
Stress-strain	Young's modulus	1.76×10^5 kg/cm^2 (2.5×10^6 lb/in^2)
	Poisson's ratio	0.18
Strength	Unconfined Compressive Strength	1340 kg/cm^2 (19,000 lb/in^2)
	Tensile Strength	0 kg/cm^2
Thermal	Coefficient of Linear Thermal Expansion	8.1×10^{-6} °C^{-1} °C^{-1} (4.5×10^{-6} °F^{-1})
	Heat Capacity	
	Temperature, °C	Heat Capacity, W•sec/kg•°C*
	0 (32°F)	880 (0.21 BTU/lb-°F)
	100 (212)	920 (0.22)
	200 (392)	960 (0.23)
	Thermal Conductivity	
	Temperature, °C	Conductivity W/m °C
	0 (32°F)	2.85 (1.65 BTU/hr-ft °F)
	50 (122)	2.70 (1.56)
	100 (212)	2.56 (1.48)
	150 (302)	2.44 (1.41)
	200 (392)	2.34 (1.35)
	300 (572)	2.15 (1.24)
	400 (752)	1.99 (1.15)
Hydraulic Conductivity	10^{-8} to 10^{-6} cm/sec	

Water/Moisture. Moisture in granite produces varying effects on its compressive strength. Based on tests[53] on two unnamed granites, each exhibited varying strength characteristics in dry and saturated conditions. For one granite a higher compressive strength was indicated when saturated than when dry. For the other granite the reverse was found. The tests were inferred to be inconclusive, but work by other investigators suggests that the change in strength is unlikely to vary more than 10% from air-dried granite specimens.

Irradiation. The effects of irradiation on the components of granite have been discussed by Jenks.[54] Study of the general aspects of radiation damage and energy storage in these materials reveals no apparent reason to expect significant impacts on the granite from irradiation.

7.2.3.3 Shale Site

Geological considerations discussed in section 7.2.1 require that the argillaceous rock formation considered in this study be placed no deeper than about 450 m (1500 ft) below ground surface. Review and comparison of the argillaceous deposits described in Section 7.2.2 indicated that these conditions could be found in several sections of the United States. The generic stratigraphic section is shown in Figure 7.2.7.

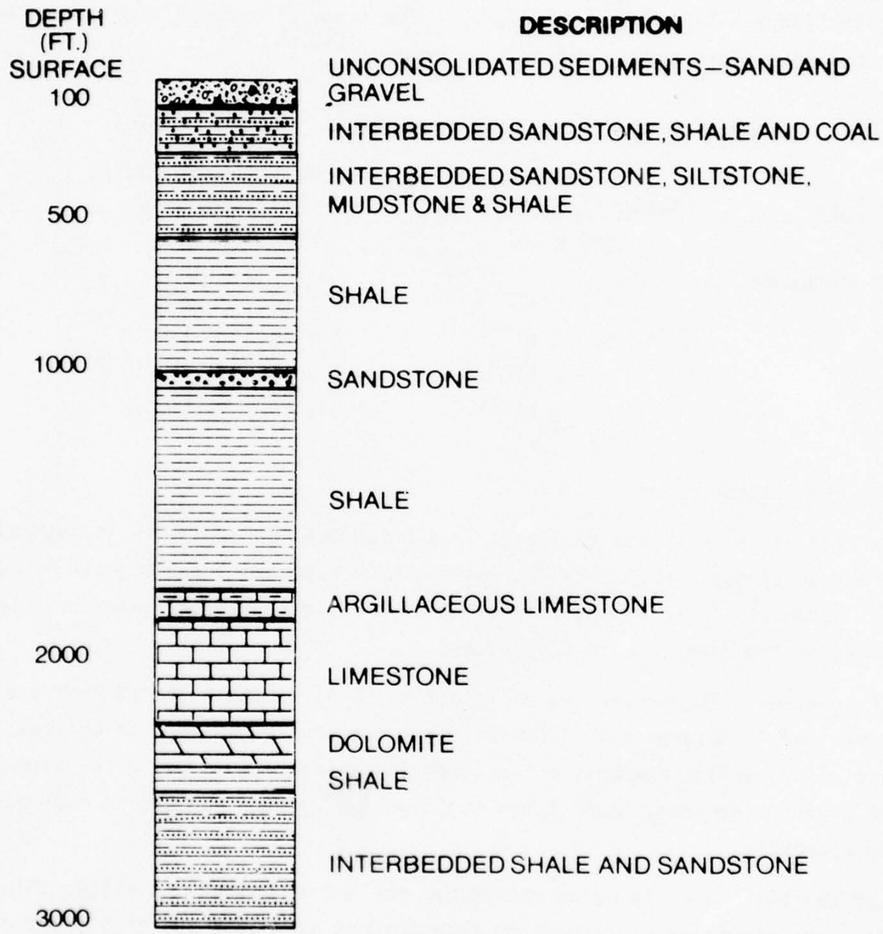

FIGURE 7.2.7. Generic Stratigraphic Section - Shale

The generic section was derived using a method similar to that used to develop the generic salt section (section 7.2.3.1). The argillaceous rock at the repository level is a marine deposited shale. It is relatively strong and, therefore, somewhat more permeable and could represent either a calcareous, clay-bonded, or carbonaceous shale. However, for purposes of this study, it will be assumed that it is not a montmorillonite rich or carbonaceous shale.

Hydrogeology. The generic shale section is presumably located near the periphery of a large, undeformed geologic sedimentary basin where the regional dip is very low and the strata are nearly flat-lying. The hydraulic gradient is very low and the potentiometric surface is about equal to the land surface. The repository host rock is overlain and underlain by various sedimentary rock units which contain and transmit groundwater at varying rates and in varying quantities depending upon the variations in permeability, porosity, and formation thickness.

The pertinent hydrogeologic parameters for the shale host rock and other formations within the generic section are listed in Table 7.2.5. The range of permeability values have been obtained from a literature survey and past experience.[50,55]

TABLE 7.2.5. Hydrologic Parameters for Generic Shale

Rock Type	Generic Hydraulic Conductivity (cm/sec)	Hydraulic Conductivity Range (cm/sec)	Porosity (%)
Sand and Gravel	6×10^{-3}	0.001 - 0.1	40
Interbedded Sandstone, Coal, Shale	1.2×10^{-3}	5×10^{-7} - 4×10^{-3}	25
Sandstone	3.5×10^{-4}	1×10^{-10} - 1×10^{-2}	20
Shale	7.1×10^{-7}	1×10^{-10} - 1×10^{-3}	16
Limestone	4.7×10^{-5}	5×10^{-8} - 1×10^{-2}	20

In-situ Rock Stresses

The vertical in-situ stress in shales in a broad sedimentary basin is typically about equal to the overburden stress. Horizontal stresses can be expected to be between 0.5 and 1.0 times the vertical stress.[52] Accordingly, a value of 1.0 times the vertical stress was considered representative of the generic site conditions.

Rock Properties. The properties of intact shale have been obtained from the literature[55] and are summarized in Reference 1. However, this summary should be used only as a rough guide to the range of properties commonly encountered in shales because of their extreme variability. They can be divided into many basic types (Section 7.2.2) each exhibiting distinctly different engineering properties.

The typical shale that is being considered for the repository is silty, thinly laminated and fissile, with occasional interbeds of fine-grained sandstone and thin seams of very stiff clay. The clay seams are depositional and limited in area. The dip of the beds is generally only a few degrees. A partial description of the typical structures expected to be found in the generic shale is contained in Reference 1.

It would be almost impossible to build a repository in shale at 1500 feet assuming average strength values for all shales. Therefore one of the more competent shales was chosen for the design of the underground repository. Estimated rock-mass property values for the generic shale are shown on Table 7.2.6. These values were estimated by methods discussed in Reference 55.

TABLE 7.2.6. Estimated Rock Mass Properties for Generic Shale

Type of Property	Parameter	Value			
Index	Unit Weight, kg/m^3	2560 (160 lb/ft^3)			
	Natural moisture content, %	2			
	Porosity (effective), %	4.0			
	Slake durability index, % (one cycle)	90			
		Horizontal[a]		Vertical[b]	
Stress-strain	Young's modulus, kg/cm^2	4.2×10^4	(6.0×10^5 $lb/in.^2$)	2.1×10^4	(3.0×10^5 $lb/in.^2$)
	Poisson's ratio	0.15		0.15	
Strength	Unconfined Compressive Strength, kg/cm^2	280	(4000 $lb/in.^2$)	280	(4000 $lb/in.^2$)
	Tensile Strength, kg/cm^2	0		0	
Thermal	Coefficient of Linear Thermal Expansion, $°C^{-1}$	8.1×10^{-6}	(4.5×10^{-6} $°F^{-1}$)	8.1×10^{-6}	(4.5×10^{-6} $°F^{-1}$)
	Heat Capacity				
	Temperature, °C	Heat Capacity, W·sec/kg·°C*			
	0 (32°F)	800 (0.19 BTU/lb-°F)			
	100 (212)	840 (0.20)			
	200 (392)	880 (0.21)			
	Conductivity				
	Temperature, °C	Conductivity, W/m °C			
	0 (32°F)	1.68 (0.97 BTU/hr-ft·°F)			
	50 (122)	1.61 (0.93)			
	100 (212)	1.54 (0.89)			
	150 (302)	1.52 (0.88)			
	200 (392)	1.51 (0.87)			
	300 (572)	1.49 (0.86)			
	400 (752)	1.47 (0.85)			

a. Horizontal = Parallel to bedding (i.e., normal to zx and zy planes.)
b. Vertical = Perpendicular to bedding (i.e., normal to xy plane).
* 1 W·sec/kg·°C = 1 J/kg·°C

Environmental Effects. The properties of shale will be affected by environmental factors.

Time (Creep). While some shales exhibit pronounced creep behavior, the deeply seated, tight shale formations, having water contents of around 2% creep less readily.[55]

Temperature. It is believed that shales become more ductile and that strengths are reduced with increases in temperature. However, no specific information on these effects has been found in the literature.

Water/Moisture. Water can cause swelling of shales, especially when montmorillonite is present.[55] Swelling clays may lead to major support problems underground, especially if the clay exposures are not immediately sealed with shotcrete. The pressures developed are considerable and are sufficient to cause buckling of steel supports.[55] For the studies described in this report, it was assumed that the shale did not contain water sensitive minerals.

Irradiation Effects. Clays and shales have high cation and anion absorption properties. However, little is known about the effects of irradiation on the physical properties of shales and further research is required.

7.2.3.4 Basalt Site

The representative stratigraphy selected for the generic waste repository in basalt consists of a series of layer flows of varying thickness, interbedded with intra-flow sedimentary deposits and overlain by alluvial cover.(56) The generic stratigraphic section is shown in Figure 7.2.8.

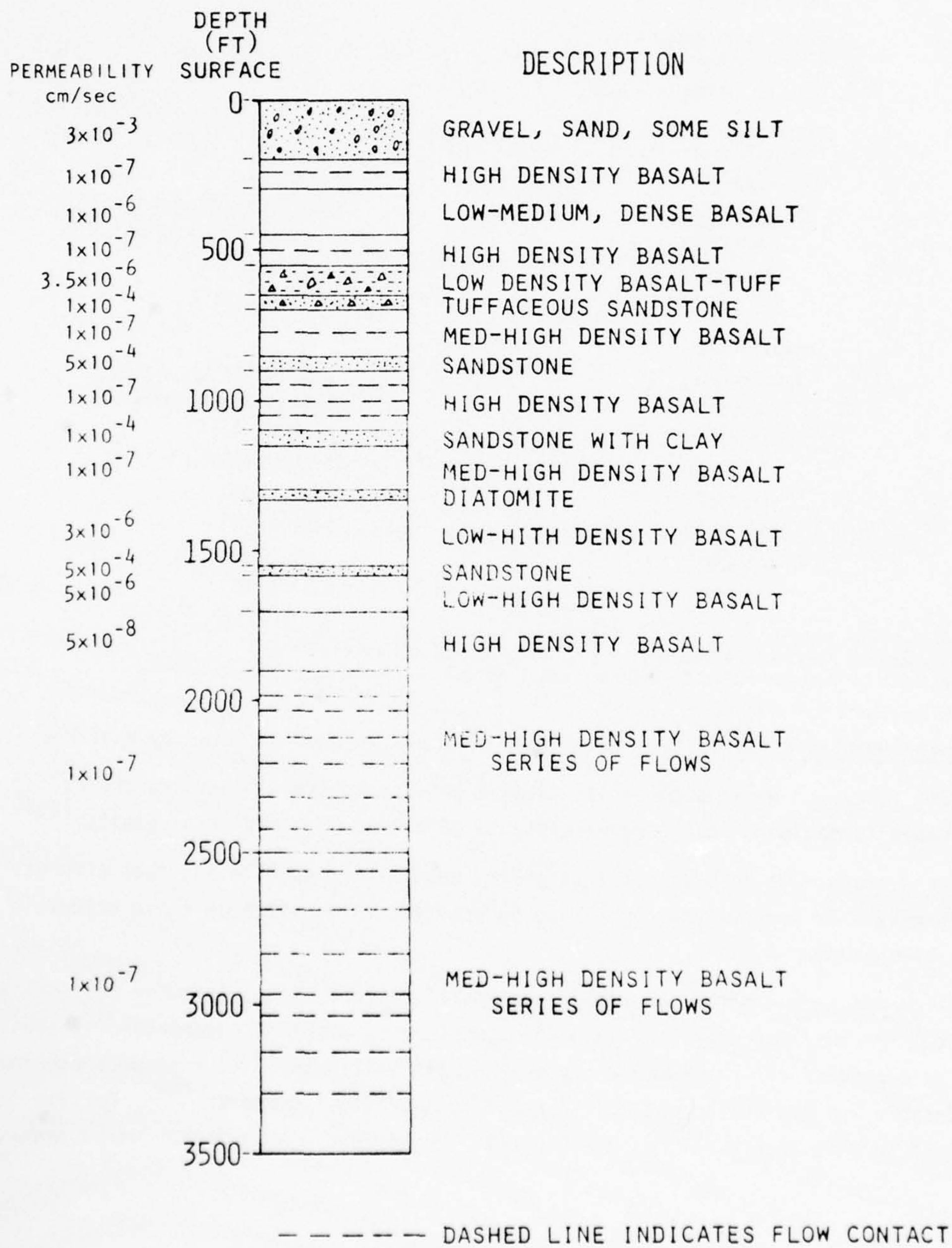

FIGURE 7.2.8. Generic Stratigraphic Section - Basalt

TABLE 7.2.6. Estimated Rock Mass Properties for Generic Shale

Type of Property	Parameter	Value			
Index	Unit Weight, kg/m^3	2560 (160 lb/ft^3)			
	Natural moisture content, %	2			
	Porosity (effective), %	4.0			
	Slake durability index, % (one cycle)	90			
		Horizontal[a]		Vertical[b]	
Stress-strain	Young's modulus, kg/cm^2	4.2×10^4	(6.0×10^5 $lb/in.^2$)	2.1×10^4	(3.0×10^5 $lb/in.^2$)
	Poisson's ratio	0.15		0.15	
Strength	Unconfined Compressive Strength, kg/cm^2	280	(4000 $lb/in.^2$)	280	(4000 $lb/in.^2$)
	Tensile Strength, kg/cm^2	0		0	
Thermal	Coefficient of Linear Thermal Expansion, $°C^{-1}$	8.1×10^{-6}	(4.5×10^{-6} $°F^{-1}$)	8.1×10^{-6}	(4.5×10^{-6} $°F^{-1}$)
	Heat Capacity				
	Temperature, °C	Heat Capacity, W·sec/kg·°C*			
	0 (32°F)	800 (0.19 BTU/lb-°F)			
	100 (212)	840 (0.20)			
	200 (392)	880 (0.21)			
	Conductivity				
	Temperature, °C	Conductivity, W/m °C			
	0 (32°F)	1.68 (0.97 BTU/hr-ft·°F)			
	50 (122)	1.61 (0.93)			
	100 (212)	1.54 (0.89)			
	150 (302)	1.52 (0.88)			
	200 (392)	1.51 (0.87)			
	300 (572)	1.49 (0.86)			
	400 (752)	1.47 (0.85)			

a. Horizontal = Parallel to bedding (i.e., normal to zx and zy planes.)
b. Vertical = Perpendicular to bedding (i.e., normal to xy plane).
* 1 W·sec/kg·°C = 1 J/kg·°C

Environmental Effects. The properties of shale will be affected by environmental factors.

Time (Creep). While some shales exhibit pronounced creep behavior, the deeply seated, tight shale formations, having water contents of around 2% creep less readily.[55]

Temperature. It is believed that shales become more ductile and that strengths are reduced with increases in temperature. However, no specific information on these effects has been found in the literature.

Water/Moisture. Water can cause swelling of shales, especially when montmorillonite is present.[55] Swelling clays may lead to major support problems underground, especially if the clay exposures are not immediately sealed with shotcrete. The pressures developed are considerable and are sufficient to cause buckling of steel supports.[55] For the studies described in this report, it was assumed that the shale did not contain water sensitive minerals.

It should be noted that the generic section shown in Figure 7.2.8 is not generic in the true sense, because it was developed primarily from information taken from the Columbia River Basalt Group. Because of its thickness, depth and extent, this formation represents the volcanic rock which would most likely meet those geologic considerations required for a waste repository site.

Hydrogeology. Columbia River Basalts often have brecciated/jointed upper and lower surfaces of each flow and may transmit considerable water. Groundwater is found in the interflow zones and is generally confined (artesian) water. The flow of groundwater occurs through the fissures and joints and/or singular disconformities along individual flows of basalt.[50]

The hydraulic conductivity (permeability) of basaltic rocks is generally low and will vary depending on the thickness, origin and cooling history. As discussed in References 50 and 56, a hydraulic conductivity of approximately 10^{-6} to 10^{-8} cm/sec was assumed to be representative for the generic site. The lower values are representative of a thick, dense flow at a depth where the discontinuities would be reduced in size and frequency due to the higher in-situ stresses.

Values of porosity for basaltic rocks[50,56] are directly related to the frequency and size of fractures, joints, and gas vesicles. Unless the rock is unusually weathered and altered by geologic or tectonic activity, it is highly unlikely that the porosity would exceed 1%. As discussed in Reference 56, a value of 0.6% was chosen for the generic basalt host rock.

In-Situ Rock Stresses. The in-situ stresses in basaltic rocks are dependent upon depth and previous stress history, particularly tectonic activity. The vertical stress is usually at least equal to the overburden pressure. For the generic conditions it has been assumed that the vertical stresses will be equal to the overburden. The horizontal stress is usually expressed as a fraction of the vertical stress; this fraction lies in the range of 0.5 to 2.0 and is generally considerably greater than 1.0 for basalt. Accordingly, a value equal to 1.5 of the vertical stress has been assumed as representative of the generic site conditions.[52]

Rock Properties. In order to obtain an idea of the properties of intact basalt a literature search[56] was carried out similar to that for the salt and the results are summarized in Reference 1.

The unit weight is fairly constant for dense basalt at about 185 lbs/ft^3, and the natural moisture content and porosity are very low. In these competent rocks Young's Modulus is high, as are the compressive and tensile strengths.[56]

Dense basalts comprise some of the more competent rocks occurring in the earth's crust. On the other hand, shear zones and intraflow erosional surfaces are frequently encountered in the rock mass and these can cause serious support and groundwater flow problems in an underground chamber.

Basalt sometimes exhibits pronounced anisotropy because of the presence of a number of small cracks. While this anisotropy affects the properties of the intact material, basalt is less likely to be anisotropic at the scale of the thick dense flow being considered as the

repository host rock. Although joint orientations may appear to be randomly oriented, definite trends are sometimes apparent. At the proposed repository depth, joints and fractures are assumed to be tight.

A partial description of structures in a typical basalt is contained in Reference 1. It is assumed that these types of structures are present in the generic basalt medium.

Based on the intact properties, the assumed structure, and a detailed review of the rock mass properties of several basalts,[50] the rock-mass property values outlined in Table 7.2.7 were selected as properties to be used in the conceptual design studies.

TABLE 7.2.7. Estimated Rock Mass Properties for Generic Basalt

Type of Property	Parameter	Value
Index	Unit Weight	2880 kg/m^3 (180 lb/ft^3)
	Natural moisture content	--
	Porosity (effective)	0.6%
Stress-strain	Young's modulus	1.3×10^5 kg/cm^2 (1.8×10^6 $lb/in.^2$)
	Poisson's ratio	0.26
Strength	Uniaxial Compressive Strength	1300 kg/cm^2 (18,000 $lb/in.^2$)
	Tensile Strength	0 kg/cm^2
Thermal	Coefficient of Linear Thermal Expansion	5.4×10^{-6} $°C^{-1}$ (3×10^{-6} $°F^{-1}$)
	Heat Capacity	
	Temperature, °C	Heat Capacity, W•sec/kg•°C*
	0 (32°F)	710 (0.17 BTU/lb-°F)
	100 (212)	800 (0.19)
	200 (392)	920 (0.22)
	300 (572)	960 (0.23)
	Thermal Conductivity	
	Temperature, °C	Conductivity W/m °C
	0 (32°F)	1.16 (0.67 BTU/hr-ft)
	50 (122)	1.19 (0.69)
	100 (212)	1.26 (0.73)
	150 (302)	1.31 (0.76)
	200 (392)	1.37 (0.79)
	300 (572)	1.49 (0.86)
	400 (752)	1.56 (0.90)
Hydraulic Conductivity	10^{-8} to 10^{-6} cm/sec	

*1 W•sec/kg•°C = 1 J/kg•°C

Environmental Effects. The properties of basalt will be affected by environmental factors.

Time. Very few tests of long duration have been carried out on basalts, and the time-dependent behavior can only be inferred from triaxial compression tests. Although a creep law for basalt has not been determined, it should be noted that the effects of creep in basalt are likely to be very small when compared to salt or shale.

Temperature. The effect of increased temperature on basalt commonly produces lower yield stresses, reduced moduli values and a reduction in Poisson's ratio.[56]

Pressure. The effect of an increased confining pressure on basalt is to increase the ultimate strength and the Young's modulus value. Poisson's ratio tends to decrease slightly with an increase in pressure.[50]

Water/Moisture. At present, no data are available for evaluating the effects of water/moisture on the properties of basalt.

Irradiation. The available information regarding radiation damage and energy storage behavior in the components of basalt has been reviewed by Jenks.[54] There is little direct experimental information for the evaluation of the effects of irradiation on the properties of basalt.

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7.3 GEOLOGIC REPOSITORY THERMAL CRITERIA

7.3 GEOLOGIC REPOSITORY THERMAL CRITERIA

A major factor associated with geologic isolation of radioactive waste is the heat generated by solidified high-level waste (SHLW) or spent fuel assemblies. This heat flows from the waste, through the emplaced canister and other protective material, into the host rock formation, through the rock surrounding or overlying this formation, and eventually out into the atmosphere. The heat will have definite impacts on:

- the integrity and recoverability of the waste canisters
- room and pillar stability
- integrity of the waste matrix over long periods of time
- integrity of the host rock and the surrounding rock
- any overlying aquifers and buoyancy effects on groundwater flow
- long-term uplift and subsidence of overlying rock.

To assure that the impact of the heat on these factors will not be detrimental to waste isolation objectives, a systematic determination of the repository design thermal loads is required and should include:

- establishment of limits for conditions affected by heat
- determination of acceptable thermal loads that will not bring about conditions beyond the assigned limits
- development of repository design thermal loads, taking into account safety, engineering, and operational requirements.

Design limits for the repository can be specified as temperature and thermomechanical criteria. Preliminary estimates of acceptable thermal conditions are summarized in Table 7.3.1. For convenience, the criteria, subsequent analyses, and results are classified into three categories: far-field, near-field, and very-near-field. The far-field refers to the geologic formation at distances far removed from the repository. The near-field is the region within the repository horizon in the vicinity of the emplacement rooms and associated pillars. The very-near-field refers to the waste package and the rock within a few feet of the canister.

The limits shown in Table 7.3.1 are based on the best available data at this time. As such, they should be reevaluated as more data become available. These limits also require evaluation on a site specific basis.

7.3.1 Acceptable Thermal Loads

The heat induced into the repository and surrounding formation depends upon repository design, which includes the thermal loadings of the repository. These loadings involve: 1) the average repository areal loading that determines the temperature rise of the formation in the far-field; 2) the local thermal loadings (average amount of waste emplaced per unit storage area of the repository) that most directly determine the near-field rock thermal and thermomechanical environments; and 3) individual canister loadings that most directly influence the

TABLE 7.3.1. Thermal and Thermomechanical Limits for Conceptual Design Studies

Event	Limits
Far-Field Considerations	
Maximum uplift over repository	1.2 to 1.5 m[1]
Temperature rise at surface	$<0.5°C$[2]
Temperature rise in aquifers	$<6°C$[2]
Near-Field Considerations	
Room closure during retrievability period - salt	$<10-15\%$ of original room opening[1]
Room stability - granite, basalt rock strength-to-stress ratio	$>2:1$ within 1.5 m of openings[3]
Room stability - shale with continuous support rock strength-to-stress ratio	$>1:1$ within 1.5 m of openings[3]
Pillar stability - non-salt strength-to-stress ratio	$>2:1$ across mid-height of pillar[3]
Very-Near-Field Considerations	
Maximum HLW temperature	
• Glass	500°C[4]
• Calcine	700°C[4]
Maximum spent fuel cladding temperature	200°C[4]
Maximum canister temperature	375°C[4]
Maximum salt temperature	250°C[1]
Maximum fracture of non-salt rock	15 cm annulus around canister[1]

temperatures in the waste, the canister, and the rock in the immediate vicinity of the waste canister, i.e., in the very-near-field. For a given repository design, acceptable loadings can be determined once appropriate temperature and thermomechanical limits have been established.

Thermal and thermomechanical analyses have been performed to determine acceptable thermal loading values for spent fuel repositories and HLW repositories in salt, granite, shale, and basalt. These studies use an iterative technique that integrates the waste and canister temperature criteria, room and pillar stability analyses, and far-field thermal and rock mass response analyses.

For isolation of HLW, the following steps were followed in the iterative analysis:

Step 1: Select thermal and thermomechanical criteria.

Step 2: Propose a conservative room and pillar design without consideration of an imposed thermal loading.

Step 3: Make near-field heat-transfer calculations to determine the areal thermal loading range of interest.

Step 4: Make very-near-field heat-transfer calculations to generate very-near-field temperature profiles as a function of areal thermal loading and canister loading.

Step 5: Make near-field rock mechanics calculations to determine areal thermal loading that assures room and pillar stability.

Step 6: Determine maximum canister load from Step 4 data for the areal thermal load from Step 5.

Step 7: Make far-field thermal and rock mechanics calculations to assure that far-field design limits are not exceeded.

If any of the tentative limits in Table 7.3.1 are exceeded in any of the above steps, the previous steps are revised and repeated until the calculational results indicate that the limits are not exceeded.

For spent fuel repository analyses, the above procedure was modified slightly. Because a single PWR or BWR spent fuel assembly is the smallest unit that can be placed in a single canister, the thermal load for a given canister was fixed, and steps 4 and 6 were not required. Steps 1 through 3 were followed by steps 5 and 7. Very-near-field heat transfer calculations were then performed to determine if canister or spent fuel temperature limits were exceeded.

The following simplifying and conservative assumptions were made for the analyses:

- Only high-level waste and spent-fuel canisters are considered.
- The entire repository is assumed to be loaded simultaneously and instantaneously.
- Thermal properties based on reasonable estimates are assumed. See Table 7.3.2.
- Heat removal capability of crushed backfill material is assumed to be 10% of intact rock (Section 7.2.3).
- The effects of stress upon thermal properties are not included.
- The presence of water is neglected in the thermal analysis.
- Heat removal from convection in storage rooms or around canisters or from water flow is generally neglected.
- Only simplified horizontal stratigraphies are assumed.
- No compaction or subsidence of the formation is considered.

The iterative procedure employed results in baseline thermal load design values for the canisters in terms of kW per canister at waste emplacement and for the loading of a repository room (local areal thermal load) in kW/acre. The canister load must be sufficiently low so that the waste and canister temperatures do not exceed the values in Table 7.3.1. The local areal thermal load must be sufficiently low so that rock mechanics analyses predict room and pillar stability throughout the readily retrievable period, also, near-field hydraulic conductivities should not be significantly increased, and long-term far-field restrictions should not be exceeded.

The near-field and far-field thermal limits discussed in the following sections are based on the heat generation of 10-year-old wastes at the time of emplacement in the repository. The actual age of the wastes at the time of emplacement ranges from a minimum of 6.5 years to

somewhat over 10 years. The near-field and far-field limits at the time of emplacement for the younger wastes could be increased without exceeding the criteria listed in Table 7.3.1. However, to simplify this criteria the near-field and far-field limits for 10-year old waste were used as a conservative requirement for all repository capacity estimates.

TABLE 7.3.2. Thermal Properties of Materials[a]

Material	Density, kg/m^3	Heat Capacity, $W \cdot sec/kg \cdot °C$	Conductivity, $W/m \cdot °C$
Vitrified HLW[5]	3,000	840	1.2
Calcined HLW[5]	1,300	840	.35
Stainless Steel 304L[6]	7,800	460	16
Carbon Steel[6]	7,800	460	45
Concrete	2,300	840	.93

a. Thermal properties of the geologic media are described in Section 7.2.3.

7.3.1.1 Very-Near-Field Thermal Criteria

In the very-near-field analyses a unit cell model that considers a single overpacked canister in a hole is employed for the calculations. Although only a small fraction of the canisters are assumed to require overpacking, the inclusion of an overpack in these analyses results in a more conservative canister loading limit. The void space between the overpack and the hole is assumed to be backfilled with 15 cm (6 in.) of crushed rock: 7.5 cm (3 in.) of backfill and 7.5 cm (3 in.) of fractured rock surrounding the hole. The radius of the unit cell corresponds to half the distance between canisters. Assumed insulating conditions at the boundary of the cylindrical unit cell effectively take into account the surrounding canisters in the infinite repository and provide conservative values for the temperatures near the canister. Upper and lower boundaries of the unit cell are sufficiently far removed from the region of interest that they have no impacts on the canister temperatures. The storage room is not modeled, but details of the canister are explicitly taken into account as shown in Figure 7.3.1.

The maximum allowable canister thermal loads generated by the very-near-field analyses are presented in Table 7.3.3. The specific thermal output of wastes arriving at the repositories increases with time as interim storage backlogs are worked off and younger wastes are shipped. In the case of HLW this would result in canister thermal outputs that exceed the thermal limits in Table 7.3.3. To prevent this, canister diameters are reduced as needed in 5 cm (2 in.) increments to reduce the canister contents and maintain the resulting canister thermal output within limits. This is not a variable with canistered spent fuel because the single fuel assembly per canister is assumed. This produces a maximum of 0.72 and 0.22 kW/can for PWRs and BWRs respectively for 6.5 year old spent fuel.

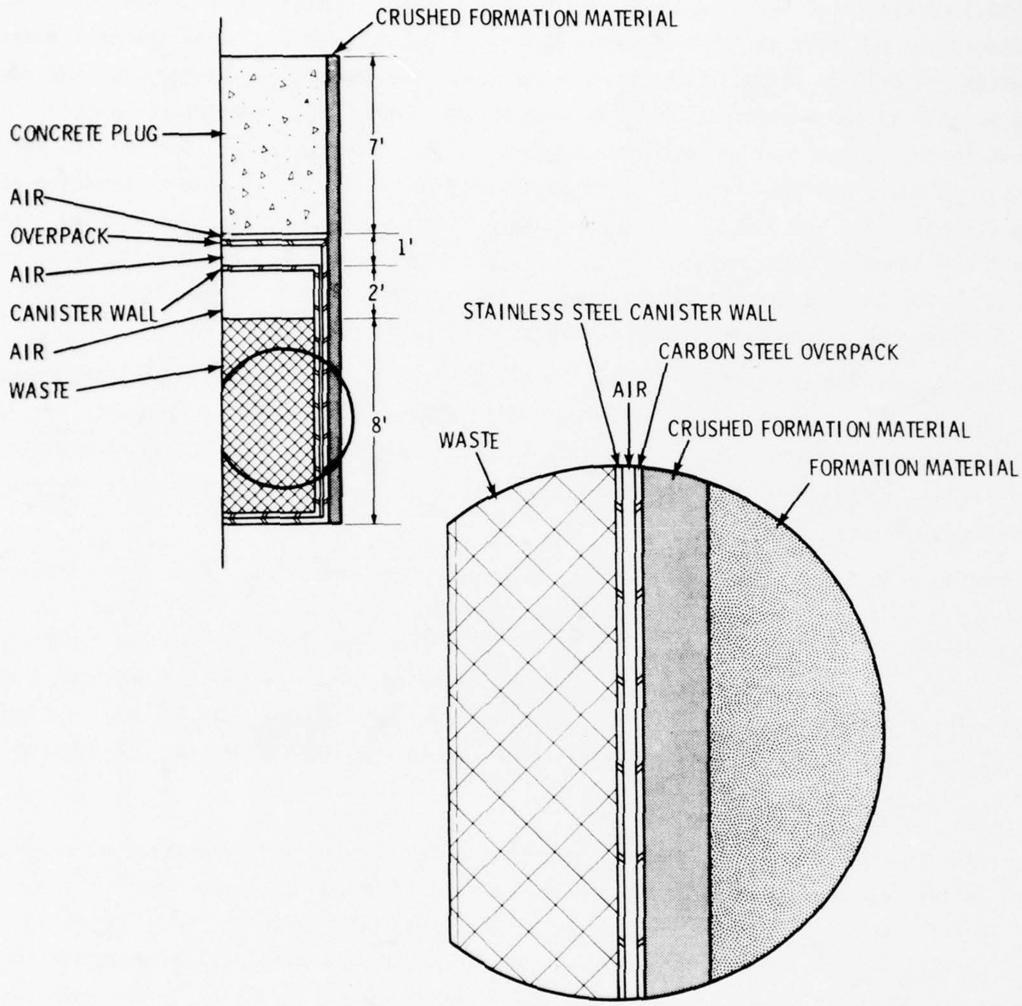

FIGURE 7.3.1. Details of HLW Very-Near-Field Unit Cell

TABLE 7.3.3. Canister Thermal Load Limits for Conceptual Repository Designs

	Thermal Load Limit, kW[a]			
	Salt	Granite	Shale	Basalt
Canister Limits During Retrieval Period[b,c]				
Vitrified glass HLW	3.2	1.7	1.2	1.3
Calcined HLW	2.6	1.6	1.1	1.1

a. Controlling factor is canister temperature limit. Limit on waste centerline temperature would allow a somewhat higher thermal load.
b. Analysis assumes 6 inch annulus of crushed rock around waste package.
c. Limit not specified for canistered spent fuel because the thermal output from a single fuel assembly is the limiting value.

Calculated time histories for temperatures in the waste (center line) or spent fuel assemblies, and in the canister wall are shown in Figures 7.3.2 through 7.3.9 for the design canister loadings and a 15 cm (6-in.) crushed rock annulus. The temperature history for the rock near the surface of the emplacement hole is also shown. These rock temperatures should correspond to the highest values obtained anywhere in the formation rock. The results for spent fuel canisters and full recycle waste canisters, respectively, in a salt formation are shown in Figures 7.3.2 and 7.3.3. The corresponding temperature histories for granite, shale, and basalt are shown in Figures 7.3.4 through 7.3.9 respectively. Temperature histories are not presented for the uranium-only recycle cases because for the 0 to 50 year time period considered here, the curves are similar to the full recycle case.

In the case of HLW in salt and in granite (Figures 7.3.3 and 7.3.5) the canister temperature limit (375°C) is exceeded for a brief period immediately following emplacement. In addition, the waste temperature (500°C) is exceeded in salt. Elimination of backfill around the canisters or backfilling with a material of higher conductivity would reduce these temperatures to within the appropriate limits.

The calculated temperature histories for PWR spent fuel in the four rock media (Figures 7.3.2, 7.3.4, 7.3.5, and 7.3.8) indicate that cladding temperatures exceed the 200°C limit. One method of reducing these temperatures is elimination of the crushed backfill surrounding the emplaced canisters. Heat is transferred across the resulting air space more readily than through the crushed backfill material and results in cooler canister and cladding temperatures. Alternatively the annulas could be backfilled with a more compact material of higher conductivity.

7.3.1.2 Near-Field Thermal Criteria

For the near-field analyses, the repository is modeled as an infinite array of storage rooms of unlimited length, with idealized waste canisters buried below the floor of the storage rooms. In this case a unit cell containing a single canister is studied. The remaining canisters and storage rooms are taken into account by assuming adiabatic (insulating) conditions at the boundaries of the unit cell. Upper and lower boundaries are sufficiently far removed that they have no impact on the storage room temperatures during the time periods of interest. Figure 7.3.10 illustrates this infinite array and unit cell.

The near-field local areal loading limits are based on room and pillar stability considerations. Near-field local loading is the thermal density for each waste type's room and pillar emplacement area excluding areas for corridors, shafts, other waste type emplacement areas, etc. Linear thermomechanical analyses based upon the predicted near-field temperature distributions indicate that readily retrievable conditions would prevail in the storage rooms for at least 5 years with the loadings in Table 7.3.4.[3]

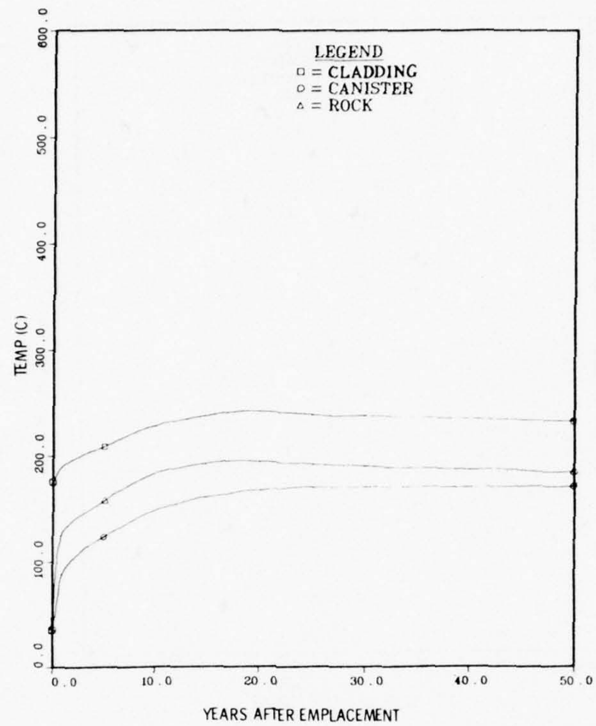

FIGURE 7.3.2. PWR Spent Fuel (50 kW/acre) Very Near Field Temperatures versus Time - Repository in Salt - Once-Through Fuel Cycle

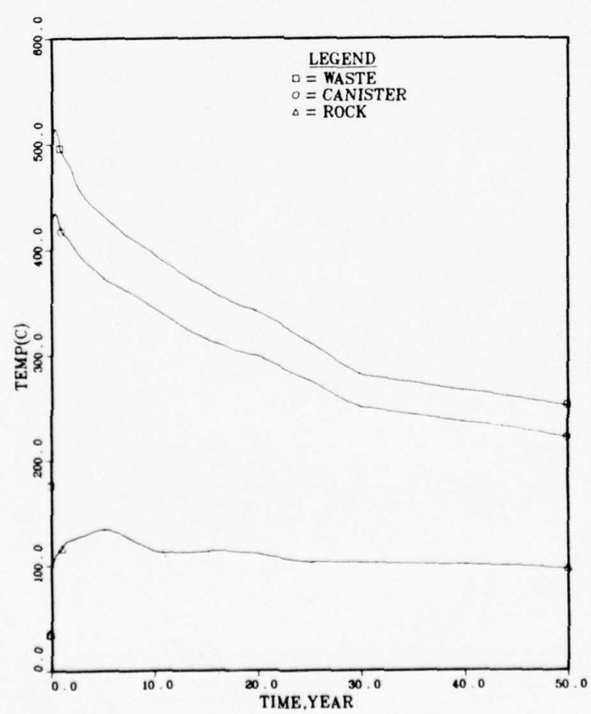

FIGURE 7.3.3. HLW Canister (100 kW/acre) Very Near Field Temperatures versus Time - Repository in Salt - Full U and Pu Recycle

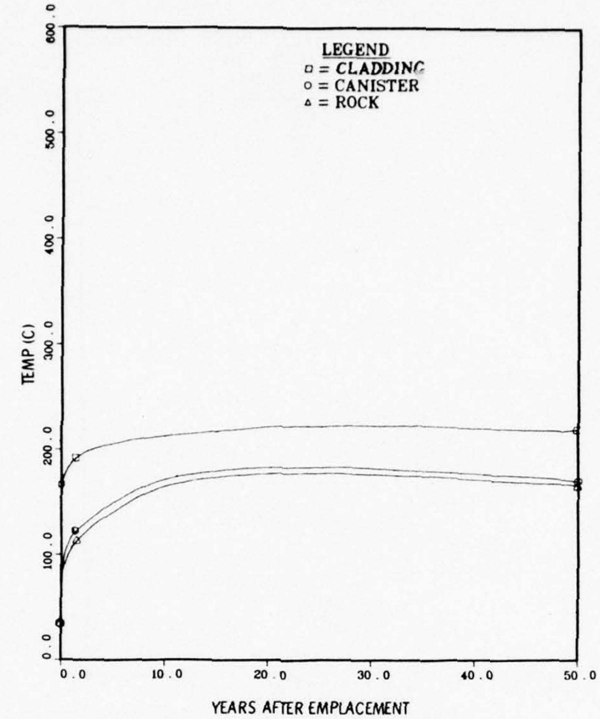

FIGURE 7.3.4. PWR Spent Fuel (130 kW/acre) Very Near Field Temperatures versus Time - Repository in Granite - Once-Through Fuel Cycle

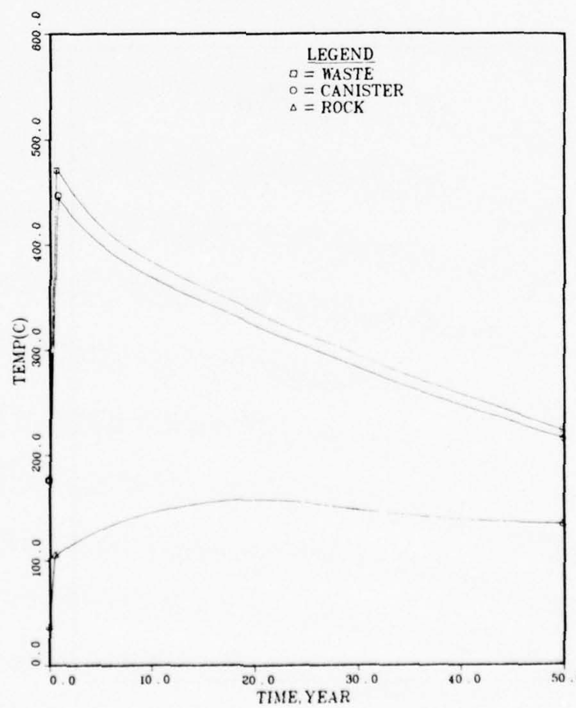

FIGURE 7.3.5. HLW Canister (130 kW/acre) Very Near Field Temperatures versus Time - Repository in Granite - Full U and Pu Recycle

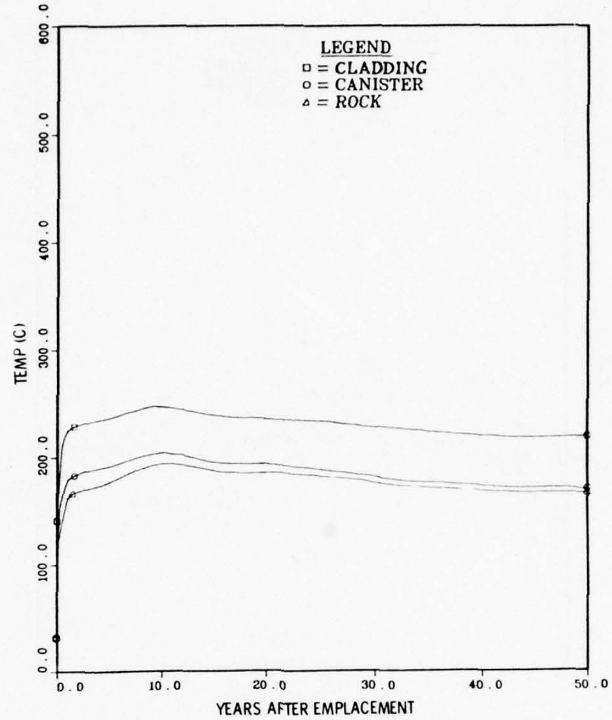

FIGURE 7.3.6. PWR Spent Fuel (80 kW/acre) Very Near Field Temperatures versus Time - Repository in Shale - Once-Through Fuel Cycle

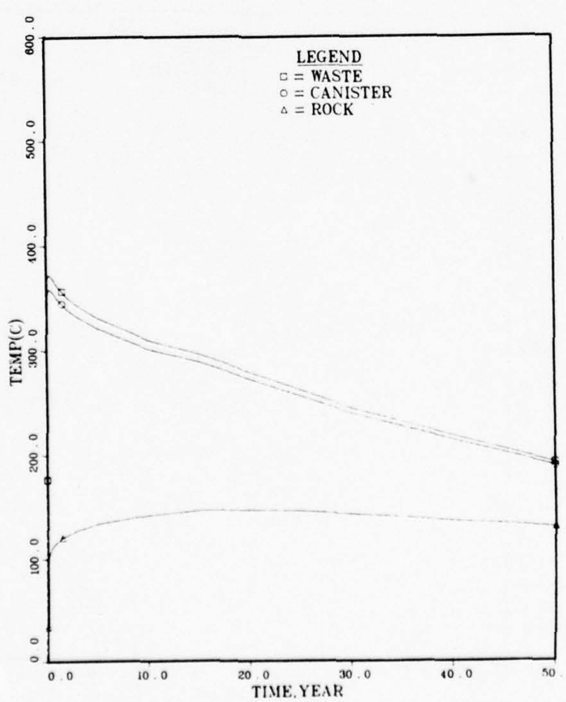

FIGURE 7.3.7. HLW Canister (80 kW/acre) Very Near Field Temperatures versus Time - Repository in Shale - Full U and Pu Recycle

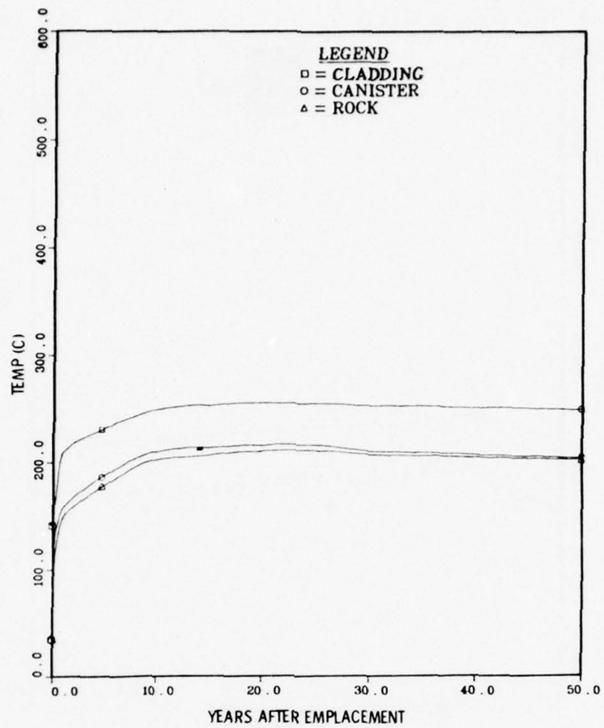

FIGURE 7.3.8. PWR Spent Fuel (130 kW/acre) Very Near Field Temperatures versus Time - Repository in Basalt - Once-Through Fuel Cycle

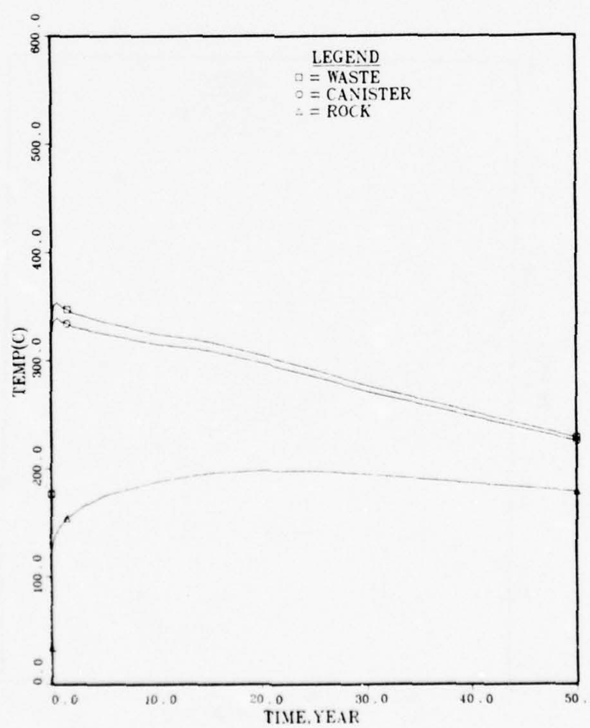

FIGURE 7.3.9. HLW Canister (130 kW/acre) Very Near Field Temperatures versus Time - Repository in Basalt - Full U and Pu Recycle

TABLE 7.3.4. Near-Field Thermal Criteria

	Thermal Load Limit[a]			
	Salt	Granite	Shale	Basalt
Near Field Local Areal Thermal Loading Limits (kW/acre)(b)				
5-year retrieval--HLW (cycles IIb and III)	150	190	120	190
5-year retrieval--spent fuel and U-only recycle, Pu in HLW	150	190[c]	120[c]	190[c,d]

a. Controlling factor is room closure.
b. Acreage includes rooms and adjacent pillars, but not corridors, buttress pillars, receiving areas, etc.
c. In order to maintain spent fuel cladding temperatures within the 200°C limit with these areal thermal loadings, the annulus around the canister is left open (no backfill). Heat is transferred across this air space more readily than through crushed backfill material and results in cooler canister and cladding temperatures.
d. The basalt repository design is based on this limit; however, later analyses indicate that the controlling factor might be the spent fuel cladding temperature and that the local areal thermal loading limit might be as low as 145 kW/acre for 5-year ready retrievability.

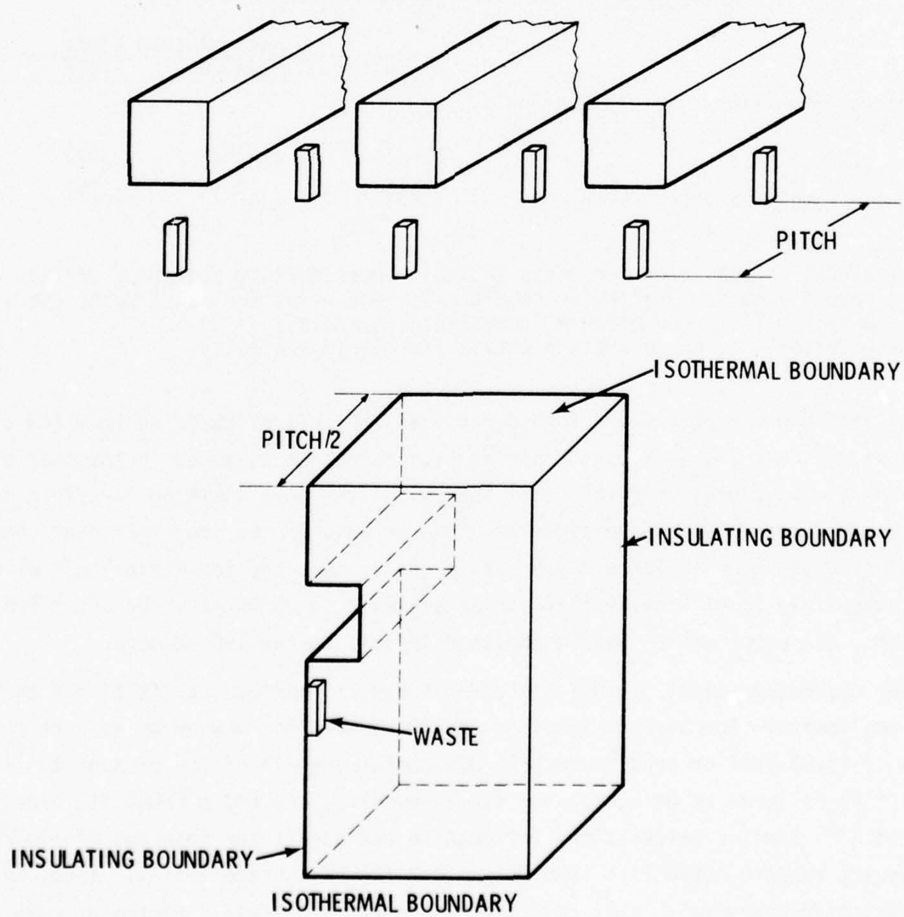

FIGURE 7.3.10. Near-Field Analyses Unit Cell

7.3.1.3 Far-Field Thermal Criteria

In the far-field model, the repository is modeled as a disc source; the heat-generation material is assumed to be homogeneously spread throughout the disc. Vertical boundaries and the lower horizontal plane boundary are removed far enough from the region of interest to have no consequence during the time span of interest. Layers of material corresponding to the generic stratigraphy are assumed to be laterally unbounded. Boundary conditions at the surface of the earth take into account an ambient geothermal flux, radiation from and to the sky, and convection.

The far-field average repository loading limits are based on the far-field studies and an estimated maximum permissible uplift of the formation that is due to heat from the stored waste. The far-field average repository loading is the thermal density for each waste type's emplacement area including corridors, ventilation drifts, etc., but does not include the areas for shafts or emplacement areas for other waste types; i.e., the thermal density for a high-heat waste section cannot be averaged with that of a low-heat waste section to meet far-field limits. Far-field thermal criteria are listed in Table 7.3.5.

TABLE 7.3.5. Far-Field Thermal Criteria

	Thermal Load Limit			
	Salt	Granite	Shale	Basalt
Far Field Average Repository Thermal Loading Limits (kW/acre)[a]				
HLW (cycles IIb and III)	150[c]	190[b]	120[b]	190[b]
Spent fuel and U-only recycle, Pu in HLW	60[c]	190[b]	120[b]	190[b]

a. Acreage includes storage area for waste including corridors, ventilation drifts, etc., but does not include area for shafts, or separate storage areas for other waste types.
b. Controlling factor is: room closure (near-field controls).
c. Controlling factor is: earth surface uplift (far-field controls).

Although salt can accept a near-field density of 150 kW/acre based on room and pillar stability considerations (Table 7.3.4), this density cannot be achieved in the case of spent fuel or HLW containing plutonium (Cycle IIa) because of the more limiting far-field criteria of 60 kW/acre. Reduced loadings are necessary here because of the long term heat contributions from the plutonium as shown in Table 7.3.6. In order to meet the far-field limit of 60 kW/acre, the maximum near-field local density that can be achieved is 75 kW/acre for spent fuel and HLW with plutonium. All other wastes may be emplaced in salt at the 150 kW/acre.

In linear thermomechanical expansion studies for salt, surface uplift of 1.2 to 1.5 m (4 to 5 ft) was obtained for average loadings of 150 kW/acre for HLW or 60 kW/acre for heat sources such as spent fuel or uranium-recycle HLW containing all of the plutonium.[1] This maximum uplift is believed to be acceptable for a repository at 600 m (2000 ft) over the time frame involved.[1] Similar calculations for granite and basalt for loadings of 190 kW/acre, and shale for 120 kW/acre, give less than 0.4 m (1.2 ft) of surface uplift. Thus, in these media, the more restrictive near-field limit restricts the far-field limit. Plutonium content does not significantly affect the non-salt media limits.

TABLE 7.3.6. Cummulative Heat Generated by 10-Year Old Spent Fuel and High Level Waste

	kW yr/MTHM	
Years	Spent Fuel Once Through Cycle	HLW U & Pu Recycle
0	0	0
10	10	9
50	34	30
100	50	36
200	72	43
300	87	46
400	99	49
500	110	50
1000	143	55

7.3.2 Thermal Loadings Achieved in Repository Designs

Engineering or operation considerations may restrict the thermal loadings to values lower than the limits presented in Table 7.3.3, 7.3.4 and 7.3.5. These considerations include such factors as reasonable HLW thermal densities, available canister sizes, structural limits on placement hole spacing, and room stability limitations on hole arrangements.

The design areal thermal loading limits for both spent fuel and HLW were conservatively set at 2/3 of the limits in Tables 7.3.4 and 7.3.5. The thermal loadings actually achieved for the conceptual repositories taking into account such things as structural limits, corridor requirements, etc., are listed in Table 7.3.7.

Temperature profiles as a function of depth that are predicted for each of the repositories using the achieved loadings are shown in Figures 7.3.11 through 7.3.18. The profiles show maximum temperatures expected in the formation at several times after the repository is loaded, and they include an ambient formation temperature effect. Temperatures shown at mine level in these figures are average values obtained throughout the mine. For example, the profiles for the spent fuel repository at a depth of 600 m (2000 ft) in salt with the average loading of Table 7.3.7 are shown in Figure 7.3.11. The figure shows that temperatures at the repository depth reach maximum values between 40 and 250 years after emplacement. The corresponding temperature increases are on the order of 28°C.

Figure 7.3.12 gives the profiles for the repository for the full recycle of uranium and plutonium. Corresponding profiles for these cycles are shown in Figures 7.3.13 and 7.3.14 for granite repositories, 7.3.15 and 7.3.16 for shale, and 7.3.17 and 7.3.18 for basalt.

7.3.3 Parametric Thermal Analyses

A series of parametric calculations were carried out to help characterize the sensitivity of very near field temperatures to variations in several parameters. Results of these parametric

TABLE 7.3.7. Thermal Loadings Achieved for the Conceptual Repositories

Cycle	Thermal Loading at Emplacement	Salt	Granite	Shale	Basalt
I	Once-Through Fuel Cycle				
	PWR				
	kW/can	0.72	0.72	0.72	0.72
	Near field local kW/acre	50	130	80	130
	Far field average kW/acre	40	100	65	100
	BWR				
	kW/can	0.22	0.22	0.22	0.22
	Near field local kW/acre	50	130	55	130
	Far field average kW/acre	40	100	44	100
IIa	U-Only Recycle, Pu in HLW				
	HLW				
	kW/can	3.2	1.7	1.2	1.3
	Near field local kW/acre	54	130	80	130
	Far field average kW/acre	40	95	60	95
	FRW				
	kW/can	0.32	0.32	0.32	0.32
	Near field local kW/acre	100	93	42	77
	Far field average kW/acre	76	70	32	60
IIb	U-Only Recycle, Pu Stored Separately				
	HLW				
	kW/can	3.2	1.7	1.2	1.3
	Near field local kW/acre	100	130	80	130
	Far field average kW/acre	76	95	60	95
	FRW				
	kW/can	0.32	0.32	0.32	0.32
	Near field local kW/acre	100	93	42	77
	Far field average kW/acre	76	70	32	60
III	Full U and Pu Recycle				
	HLW				
	kW/can	3.2	1.7	1.2	1.3
	Near field local kW/acre	100	130	80	130
	Far field average kW/acre	76	95	60	95
	FRW				
	kW/can	0.32	0.32	0.32	0.32
	Near field local kW/acre	100	93	42	77
	Far field average kW/acre	76	70	32	60

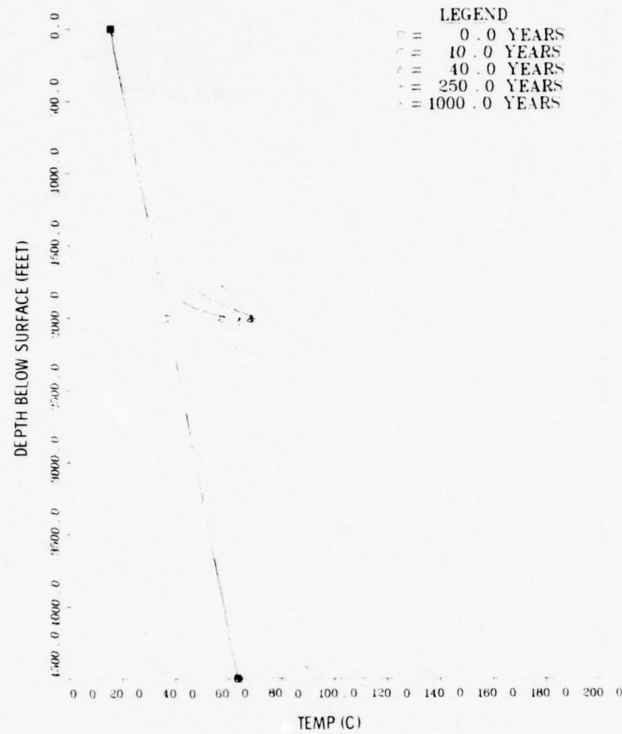

FIGURE 7.3.11. Formation Temperature versus Depth and Time - Repository in Salt - Once-Through Fuel Cycle (40 kW/acre)

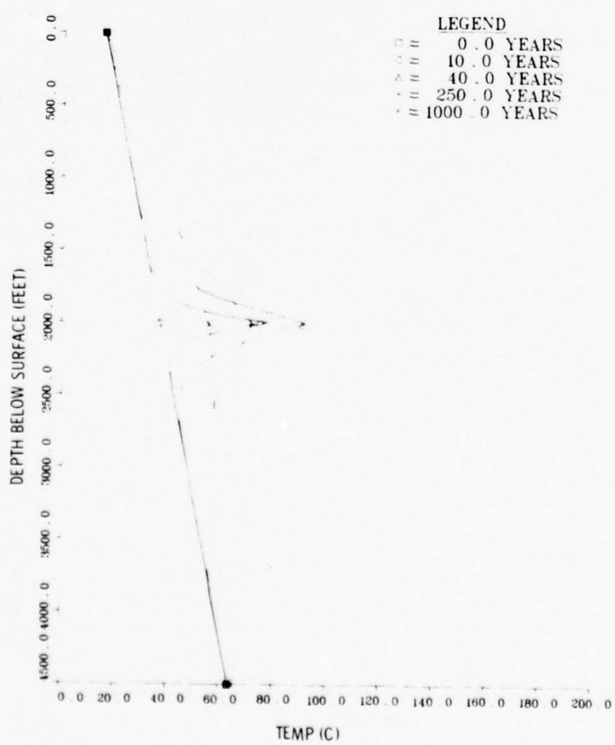

FIGURE 7.3.12. Formation Temperature versus Depth and Time - Repository in Salt - Full U and Pu Recycle (76 kW/acre)

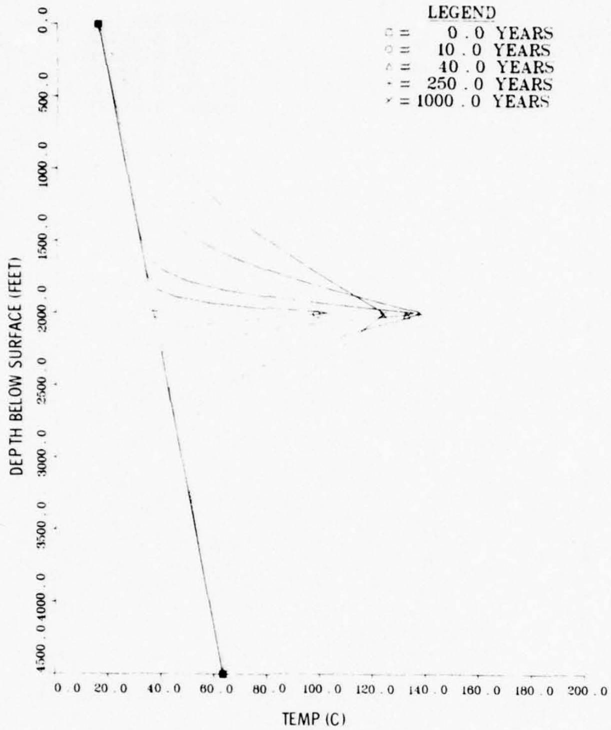

FIGURE 7.3.13. Formation Temperature versus Depth and Time - Repository in Granite - Once-Through Fuel Cycle (100 kW/acre)

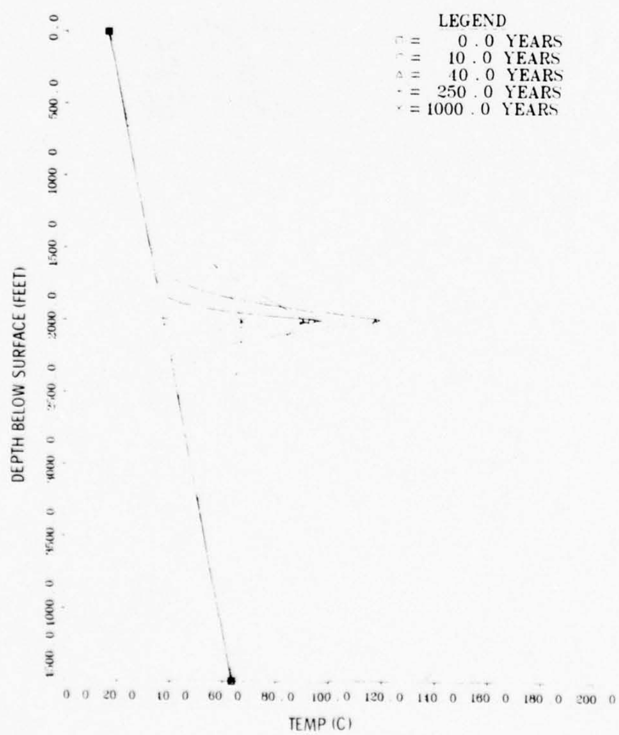

FIGURE 7.3.14. Formation Temperature versus Depth and Time - Repository in Granite - Full U and Pu Recycle (95 kW/acre)

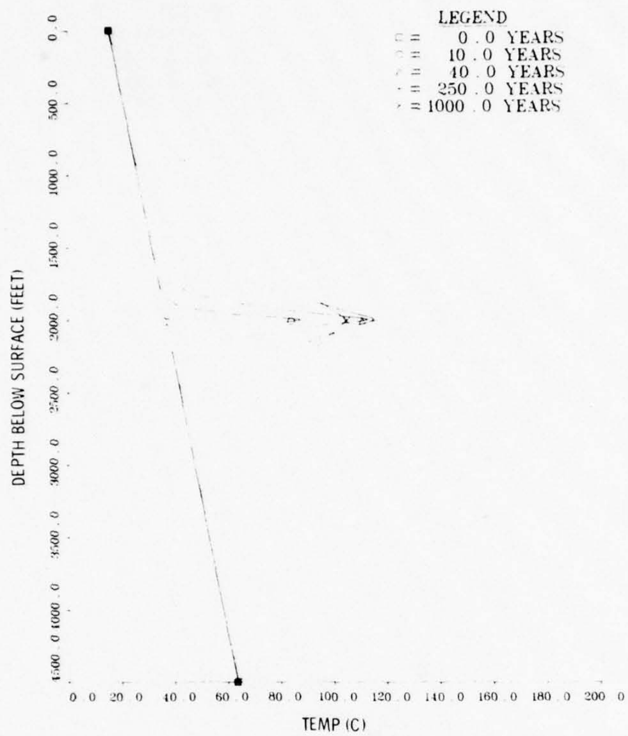

FIGURE 7.3.15. Formation Temperature versus Depth and Time - Repository in Shale - Once-Through Fuel Cycle (65 kW/acre)

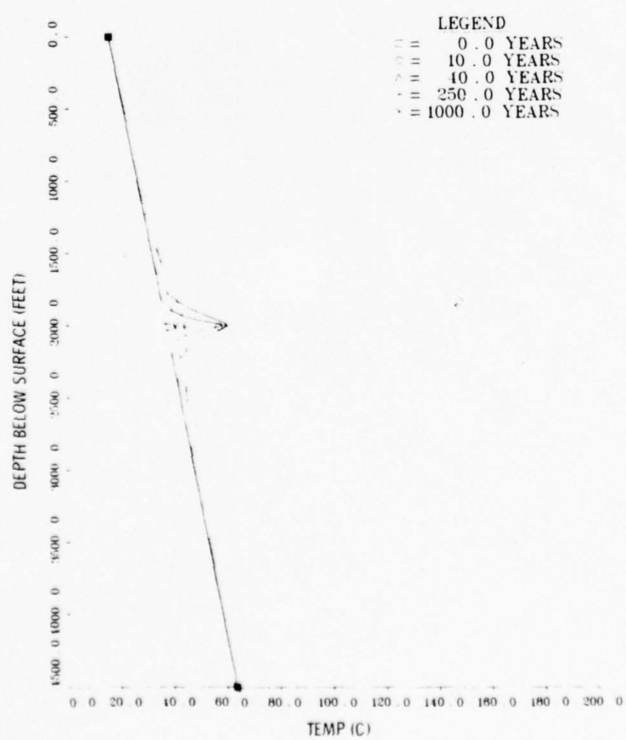

FIGURE 7.3.16. Formation Temperature versus Depth and Time - Repository in Shale - Full U and Pu Recycle (60 kW/acre)

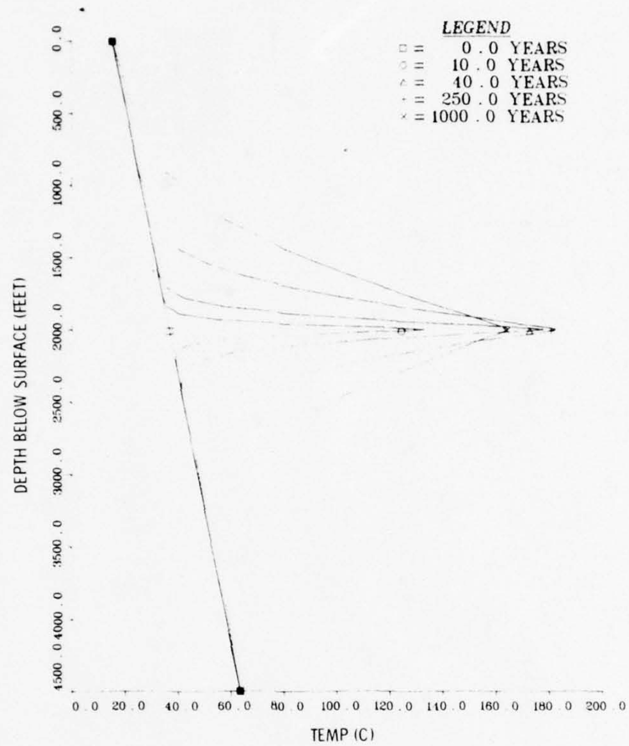

FIGURE 7.3.17. Formation Temperature versus Depth and Time - Repository in Basalt - Once-Through Fuel Cycle (100 kW/acre)

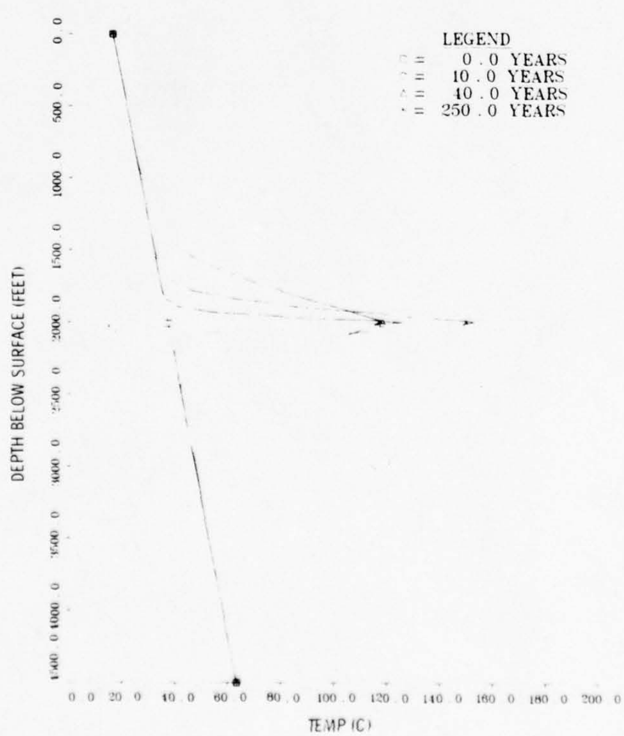

FIGURE 7.3.18. Formation Temperature versus Depth and Time - Repository in Basalt - Full U and Pu Recycle (95 kW/acre)

calculations are briefly summarized in this section. The sensitivity of spent fuel cladding, solidified high-level waste canister wall, and rock media temperatures to the following parameters were investigated:

canister heat rate, i.e. kW/canister
areal heat loading, i.e. kW/acre
canister diameter
backfill thickness
backfill conductivity
waste conductivity.

This discussion of thermal effects is in terms of the thermal limits for spent fuel and vitrified HLW defined in Tables 7.3.1 and 7.3.3 through 7.3.5.

The calculations determined the maximum temperatures achieved in the fuel cladding, in the waste, in the waste canister and in the rock. In general, the maximum cladding, canister, and rock temperatures for spent fuel are reached after 10-20 years. However, in the case of the higher heat generation rates with the HLW canisters, the maximum waste and canister temperatures are usually achieved within the first year after emplacement and then steadily decline. The maximum rock temperatures are achieved 10-20 years after emplacement.

7.3.3.1 Spent Fuel Cladding Temperatures

Spent fuel cladding temperatures are relatively insensitive to areal loading although they do increase gradually as the areal loading increases. Eliminating the crushed rock backfill around the canisters either by providing a close fit between canister and rock or leaving a gap for radiant heat transfer reduces the cladding temperature about 25-50°C in all four media. Because spent fuel canisters contents were fixed at one assembly per canister, no parameteric calculations for canister heat rate or canister size were carried out.

7.3.3.2 Effects of HLW Areal Heat Load

Waste center line temperatures are quite insensitive to areal thermal loading. Areal thermal loads could be increased more than two-fold over those used for the conceptual repositories before the thermal load limit on canisters in Table 7.3.3 would have to be reduced to maintain a 500°C maximum center line temperature.

The canister temperature is not as insensitive to areal thermal loading as the waste temperature. However, there is no problem in achieving the 375°C maximum for areal thermal load used as the design basis if 12-inch diameter canisters are used.

Rock temperature is quite sensitive to areal thermal loading but there is no problem in achieving a 250°C maximum rock temperature in all media.

7.3.3.3 Effects of HLW Canister Diameter

Waste center line temperatures increase as canister diameter is reduced, but there is no problem in maintaining a 500°C maximum center line temperature over the range of canister sizes used in the conceptual repositories.

Canister temperatures also increase as canister diameter is reduced. Only the 12" canister meets the 375°C maximum canister temperature limit in all media. A higher conductivity backfill material could be used to reduce the canister temperature if this limit is critical for smaller diameter canisters.

The maximum temperature of the surrounding rock is quite insensitive to variations in canister diameter and there appears to be no problem in maintaining a maximum 250°C temperature in all media.

7.3.3.4 Effects of Backfill Zone Thickness

Waste temperature and canister temperatures are sensitive to the thickness of the crushed rock backfill around the canister. The sensitivity is greatest in salt and least in basalt. For example, a reduction of the backfill thickness from 15 cm (6 in.) to 10 cm (4 in.) reduces the waste temperature about 70°C in salt and about 20°C in basalt. The effect on canister temperatures is approximately the same magnitude.

The maximum rock temperature is relatively insensitive to the thickness of backfill.

Effects of Backfill Conductivity

Waste and canister temperatures are also sensitive to the conductivity of the backfill. A 20% change in the backfill conductivity can result in a change of 50°C or more in both waste and canister temperatures.

Rock temperatures are quite insensitive to backfill conductivity.

Effects of Waste Conductivity

Increases in waste conductivity show little effect on waste centerline temperatures. However, decreases in conductivity can substantially increase waste temperatures.

Neither canister temperatures nor rock temperatures are much affected by wide variations in the conductivity of the waste.

Conclusion

Waste canister emplacement configurations can be designed to accomodate critical very near-field temperature constraints over a broad range of temperatures. If, in the interest of long term waste stability and integrity it should prove desirable to reduce the canister temperature below the 375°C limit assumed here, several methods are available to accomplish this. For example, reduced waste content per canister, larger diameter canisters and special backfill material with good heat conductivity properties are all possibilities. Maximum canister temperatures of 300°C could be achieved relatively easily in all four geologic media considered here and maximums as low as 200°C appear feasible. Of the four media considered, these temperatures could be achieved most easily in a salt medium and would be the most difficult in basalt because of the relative conductivity of these materials. This lower maximum would, however, only be achieved at the expense of increased disposal costs.

REFERENCES FOR SECTION 7.3

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7.4 GEOLOGIC REPOSITORIES FOR THE ONCE-THROUGH FUEL CYCLE

7.4 GEOLOGIC REPOSITORIES FOR THE ONCE-THROUGH FUEL CYCLE

Repositories operating in the once-through fuel cycle receive canistered PWR and BWR spent fuel elements. Surface facilities receive and prepare the canisters, and subsurface facilities transport and emplace them. The following sections describe conceptual repositories located in salt, granite, shale and basalt formations for the once-through fuel cycle. These descriptions are based on conceptual repositories described in References 1 and 2 modified to accommodate the waste forms described in this report. These concepts do not necessarily represent an optimum design but are representative of what could be achieved with current technology. In actual applications it is reasonable to expect that there could be some improvement over these concepts that might be reflected in either more efficient operation or lower environmental impacts or both. These conceptual descriptions provide a reasonable basis for cost analysis and for development of environmental impacts.

7.4.1 Alternatives for Once-Through Fuel Cycle Repositories

The principal alternatives for the conceptual repository design, which are presented in the following sections, relate to the waste emplacement area, waste emplacement methods, mined room configurations, repository depth, and disposal methods for the excess mined rock.

The mine layouts at the repositories in salt, granite, shale and basalt are conventional room and pillar arrangements. In these configurations the emplacement rooms are separated by continuous pillars of intact rock, which provide ceiling and wall support. In addition, the ordered arrangement of rooms improves utilization of emplacement area. Alternatives such as cavern-type excavations or a storage tunnel arrangement would not result in as efficient use of the rock formations for waste emplacement and for structural support. Also, overall repository size (in the absence of rock formation constraints) may be increased or decreased. Smaller repositories are not as economically efficient because many repository costs are constant, regardless of repository size. Therefore, smaller repositories would have higher unit costs. However, as the repository becomes larger, underground operations become less efficient because of longer travel distances for the waste transporters and other equipment. The conceptual repository size is somewhat arbitrary but is believed to represent a feasible and reasonably efficient size. An actual repository may be either larger or smaller depending on conditions at the specific site and other factors.

For the repositories conceptualized in this report, canistered spent fuel assemblies are emplaced vertically in holes or trenches that are drilled or excavated in the floors of the emplacement rooms. This method of emplacement provides for ease of handling and simplifies retrievability should that be required. Alternatives to vertical emplacement in the floors include horizontal or angled emplacement in the pillars, and emplacement in free standing steel racks. However, these alternatives have handling, emplacement, and retrieval problems that are sufficient to preclude their consideration for these conceptual facilities.

Several methods are available for managing the excess mined rock that is not used for backfilling. These methods include:

- onsite surface storage
- commercial use
- disposal in abandoned mines
- ocean disposal.

Additional methods that apply specifically to rock salt are deep well injection and disposal in existing salt flats. Onsite surface storage was chosen for this conceptual design because the other alternatives depend heavily on repository location and other site-specific considerations. Excess mined rock disposal is discussed in Section 7.4.4 and more detailed descriptions of the excess rock disposal alternatives are provided in Reference 3.

7.4.2 Design Basis for Once-Through Fuel Cycle Repositories

The design basis for the conceptual spent fuel repositories was as follows:

- The conceptual repositories are designed to receive and emplace those packaged PWR and BWR spent fuel assemblies whose characteristics are as described in Sections 3.3 and 5.7. Repository layouts and capacities are based on the thermal criteria described in Section 7.3 and on reference 6.5-years-old (time out of reactor) spent fuel although the initial fuel receipts are older because of an accumulated backlog by the time the first repository is available.

- All spent fuel elements are packaged individually in steel canisters and are received at the repository exclusively by rail.

- The allowable thermal density for emplacement of wastes is conservatively set at two-thirds of the calculated maximum acceptable density. Additional details on thermal criteria are provided in Section 7.3.

- Overall underground repository area in salt, granite, shale, and basalt is approximately 800 ha (2000 acres). Maintaining the same size repository in each rock media, however, results in different repository capacities (total MTHM) for each media because of different thermal criteria for emplacing the spent fuel (Section 7.3).

- The repositories operate in a readily retrievable mode for the first 5 years. During this period of ready retrievability, rooms are left open (no backfill), and all emplacement holes and trenches are provided with steel liner sleeves and concrete plugs. Emplacement rooms and pillars are designed to accommodate the affects of decay heat from the spent fuel during this period and remain open and accessible. In salt, accelerated creep closure of the rooms is taken into account.

- All mining operations are completed during the 5-year readily retrievable period. This insures that the entire formation required for the repository will have been explored and found to be satisfactory during the readily retrievable period.

- Tests to confirm predicted thermal characteristics of the formation will also be completed during the 5-year readily retrievable period.

- Surface spent fuel handling facilities are designed to accommodate the canister receiving rates shown in Table 7.4.1.

TABLE 7.4.1. Spent Fuel Annual Receiving Rates

Year	Canisters PWR	Canisters BWR	MTHM PWR	MTHM BWR
1985	830	1,700	380	320
1990	3,600	5,500	1,700	1,000
1995	5,000	7,200	2,300	1,400
2000	7,500	11,000	3,500	2,100
2005	11,000	16,000	5,100	3,000
Final Year of Repository Operation				
Repository in Salt				
2000	7,500	11,000	3,500	2,100
Repository in Granite				
2009	12,000	18,000	5,500	3,400
Repository in Shale				
2002	9,100	14,000	4,200	2,600
Repository in Basalt				
2009	12,000	18,000	5,500	3,400

7.4.3 Operations at a Repository for Spent Fuel

Canistered spent fuel elements arrive by rail in shipping casks designed for fuel transport and are received at the repository's surface facilities. The casks are lifted by crane from the rail cars and are moved to cask inspection stations where they are checked for external contamination. At this point a test is made for leaking canisters by purging and testing the cask atmosphere for helium. (Because all spent fuel canisters contain a 5 psig helium atmosphere, a cask with a leaking canister will contain escaped helium). Casks containing a leaking canister are moved to the overpack cell where the leaking canister is identified and placed into an overpack canister that is then welded shut.

If leaking canisters are not indicated, the casks are lowered into a cask transfer gallery where the casks are opened. The canisters are lifted into the transfer cell and placed in a storage rack. Here the canisters are again checked for external contamination and leaks. Damaged or leaking canisters are returned to the cask for transfer to the overpack cell.

Undamaged and overpacked canisters are then transferred to the canistered waste shaft and lowered four at a time into the repository. All spent fuel handling is done remotely.

The spent fuel canisters are received at subsurface transfer stations. A shielded transporter is used to remove two of the canisters from the transfer station and deliver them to an emplacement area.

At repositories located in salt and shale formations, individual PWR and BWR canisters are lowered into vertical holes drilled into the floors of emplacement rooms. The number of canisters that can be placed in a single room is limited by the minimum hole spacing of 1.8 m (6 ft) center to center or the allowable thermal density (kW/acre), whichever is more restrictive. The minimum hole spacing limit is based on local rock strength. The thermal density, calculated by summing the thermal output of the wastes (kW/canister) over the room and pillar area, is limited by room and pillar stability or by thermal expansion limits for the formation above the repository.

In repositories located in granite and basalt, PWR canisters are emplaced in holes as described for the salt and shale repositories. However, BWR canisters are placed in sleeves in backfilled trenches that run the length of the emplacement rooms. Steel racks are used in the trenches to maintain the sleeve spacing prior to backfilling. The storage rack arrangement allows a minimum 0.9 m (3 ft) center-to-center spacing transversely within the trench and 0.9 m (3 ft) longitudially. This decreased minimum spacing permits waste emplacement that more closely approaches the allowable thermal density for these formations (see Section 7.3).

Details of hole and/or trench spacing and repository area requirements are provided in Table 7.4.2 for repositories located in salt, granite, shale and basalt formations.

TABLE 7.4.2. Repository Area Allocations and Arrangement

	Salt		Granite		Shale		Basalt	
	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR
Room size (m)[a]	5.5 x 6.7 x 1070	5.5 x 6.7 x 1070	5.5 x 7.6 x 170	5.5 x 7.6 x 170	5.5 x 7.6 x 170	7.9 x 7.6 x 170	5.5 x 7.6 x 170	5.5 x 7.6 x 170
Number of rooms	195	90	1570	720	780	480	1570	720
Canisters[b] per room	350	1156	103	340	111	273	103	340
Center to center canister spacing (m)	5.2, two rows[e]	1.8, two rows[e]	3, two rows[e]	0.9, two rows[e]	3, two rows[e]	1.8, three rows[e]	3, two rows[e]	0.9, two rows[e]
Pillar width (m)	14	14	7.6	7.6	18	18	8.2	8.2
Room and pillar area (ha)	390	180	380	180	310	210	380	180
Total emplacement[c] areas (ha)	490	230	470	220	390	260	470	220
Total[d] repository area per 1000 MTHM (ha)	17.4	12.9	7.2	5.5	12.0	13.1	7.2	5.5
Wtd. Avg.	15.7		6.6		12.4		6.6	

a. Width by height by length.
b. Canisters are not emplaced in the first 10 m of the rooms.
c. Includes area of corridors within emplacement areas.
d. Amount of total repository area including shaft maintenance, corridor, and emplacement areas required for emplacement of 1000 MTHM of spent fuel. Total area for a single repository (as conceptualized in this report) is 800 ha (2000 acres).
e. Canisters emplaced in holes. Rows are 1.8 m apart center to center.
f. Canisters emplaced in trenches 1.7 m wide by 7.6 m deep. Rows are 0.9 m apart center to center.

Emplacement holes are drilled with truck mounted bucket drilling rigs at a rate that keeps pace with the canister receiving rate. The trenches in granite and basalt are excavated using conventional drill and blast techniques. Racks containing emplacement sleeves are placed into the trenches after which the trenches are backfilled with crushed rock. The sleeves extend from the bottom of the trench to the floor of the room as shown in Figure 7.4.1 and contain the emplaced BWR canisters while the crushed backfill provides shielding. Excess crushed rock from drilling and trenching operations is removed by load-haul-dump units and transported to underground storage silos. The rock in these silos is used for backfill or is removed to the surface.

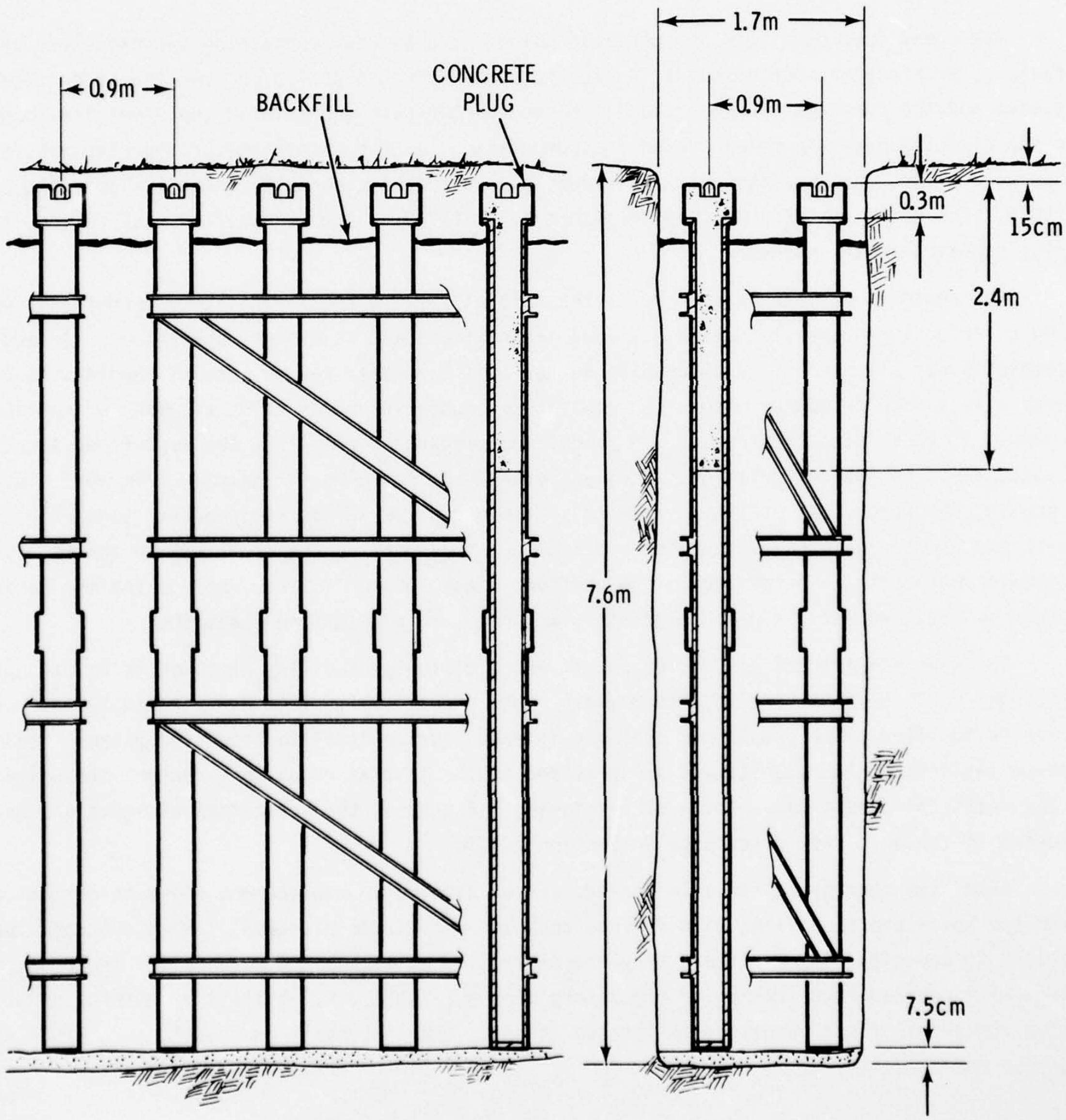

FIGURE 7.4.1. BWR Emplacement Trenches - Once-Through Fuel Cycle Repositories in Granite and Basalt

Initially, the repository is operated in a readily retrievable mode. This means that all wastes could be removed from the repository at about the same rate and with about the same effort as for emplacement. To assure canister retrievability, corrosion or other damage to waste canisters must be prevented. Canisters placed into holes during this readily retrievable period are protected by lining the holes with steel sleeves while canisters in trenches at the granite and basalt repositories are always placed in storage racks with sleeves. These sleeves encase the spent fuel canisters and prevent contact between the canisters and potentially corrosive materials and insure easy retrieval should that be necessary. Concrete plugs are used to seal the sleeves and to provide shielding. Emplacement of canisters in sleeved holes is illustrated in Figure 7.4.2.

The spent fuel canisters are emplaced using a transporter containing two canisters of spent fuel. A transporter mounted hoist is used to remove the concrete plug from the hole or trench sleeve and the operator then positions the transporter cask (containing the spent fuel canisters) directly over the hole. Using instrumentation located within the transporter cab, the operator plumbs the canister over the sleeve and activates a canister hoist to lower the canister. The hoist is then remotely unlatched and withdrawn, the transporter cask raised, and the plug replaced in the sleeve.

The readily retrievable mode spans the initial five years of repository operations, providing a period for observation of waste-rock interactions and repository operations. In addition, completion of all mining and excavation during this five-year period permits examination of the entire host rock formation within the repository boundaries prior to backfilling of emplacement rooms. If it is concluded during the readily-retrievable period that the repository location is unacceptable for waste isolation, retrieval operations would be initiated. This would involve removing the spent fuel canisters from their sleeved holes in the emplacement rooms with the same transporter originally used for emplacement, transporting the canisters to the receiving stations where they are hoisted to the surface. Some form of interim storage for the canisters would be provided until another repository was ready to receive the spent fuel.

The time required to provide an onsite interim storage facility (assumed to be dry caisson storage) could take as long as 2 to 3 years. The repository surface handling equipment must also be modified to be compatible with the interim storage facility handling systems. Unless these facilities and capabilities are provided in the initial repository design, the delay could slow retrieval operations. Costs of retrieving and storing the spent fuel elements are presented in the unit cost discussion in Section 7.4.10.

After the readily retrievable period, use of sleeves in emplacement holes is discontinued and the holes are backfilled with crushed rock after canister placement. When the rooms become filled to capacity with canisters they are backfilled to within 0.6 m (2 ft) of the room ceilings with crushed rock from prior repository mining. Table 7.4.3 lists the contents of alternative conceptual first repositories located in salt, granite, shale and basalt formations at the end of emplacement.

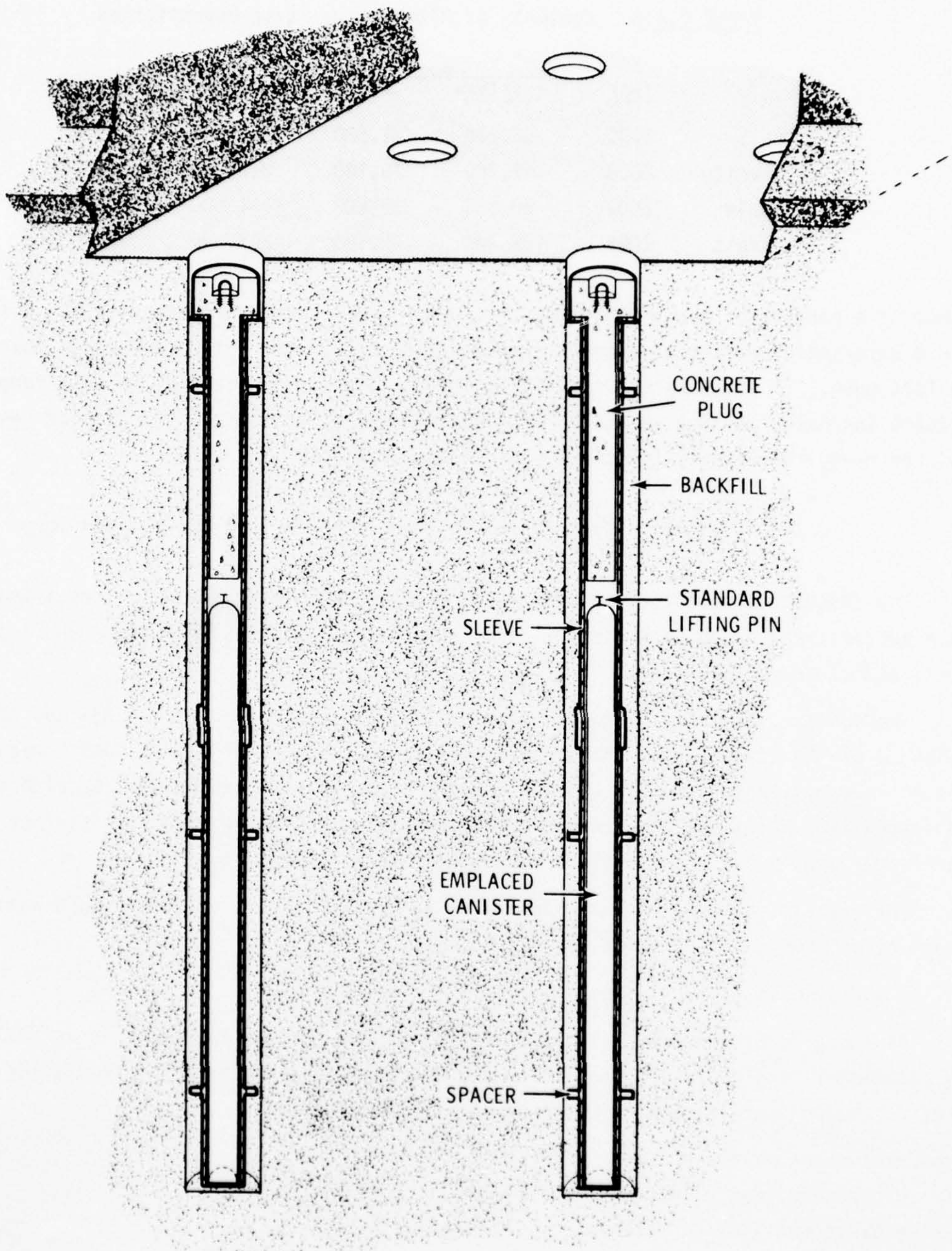

FIGURE 7.4.2. Emplacement of Canistered Spent Fuel in Sleeved Holes

Should a decision be made to extend the period of readily retrievable emplacement beyond five years, wastes would continue to require emplacement with sleeves, and backfill of rooms would be delayed. For extensions beyond a few years, the maximum areal thermal density may need to be decreased to ensure room and pillar stability for the longer retrievable periods. The areas previously filled with wastes would need to have their thermal densities decreased by

TABLE 7.4.3. Contents of Alternative First Repositories

	Year	PWR Canisters	PWR MTHM	BWR Canisters	BWR MTHM
Salt	2000	68,200	31,500	104,000	19,600
Granite	2009	162,700	75,100	246,300	46,500
Shale	2002	86,300	39,800	131,000	24,700
Basalt	2009	162,700	75,100	246,300	46,500

removing a portion of the emplaced wastes. Emplacement rooms at repositories in salt formations would experience additional creep closure as a result of the extended periods that rooms would be left open. To compensate for this increased closure, repositories in salt formations may require increased ceiling height in emplacement rooms. A brief description of requirements for 25 year ready retrievability is provided in Appendix 7.D.

7.4.4 Facility Description for Once-Through Fuel Cycle Repository

The conceptual repositories consist of surface facilities for waste receiving and handling, mine support, and mined rock storage, and subsurface facilities for waste handling and emplacement, and mined rock removal.

The surface facilities shown in the facility plot plan (Figure 7.4.3), are the only visible evidence of the repository. These facilities occupy an area of 180 ha (440 acres) at the salt and shale repositories and 280 ha (700 acres) at the granite and basalt repositories. The additional 100 ha at the granite and basalt repositories are required for surface storage of the larger amounts of rock that are mined from these formations.

The overall subsurface areas consisting of service areas, corridors, and waste emplacement rooms occupy 800 ha (2000 acres) in all cases.

7.4.4.1 Surface Facilities

As shown in Figure 7.4.3, all surface facilities are surrounded by an agricultural fence. An additional double security fence is provided for the waste handling facilities and repository shafts. Major facilities within the security fence include:

- canistered waste building
- radioactive waste treatment building
- exhaust ventilation building
- supply ventilation building
- men and material building.

Repository surface facilities designed to Category I* requirements are the canistered waste building, radioactive waste treatment building, exhaust ventilation building, supply ventilation

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* Designed to withstand maximum credible natural disasters, such as earthquakes and tornadoes.

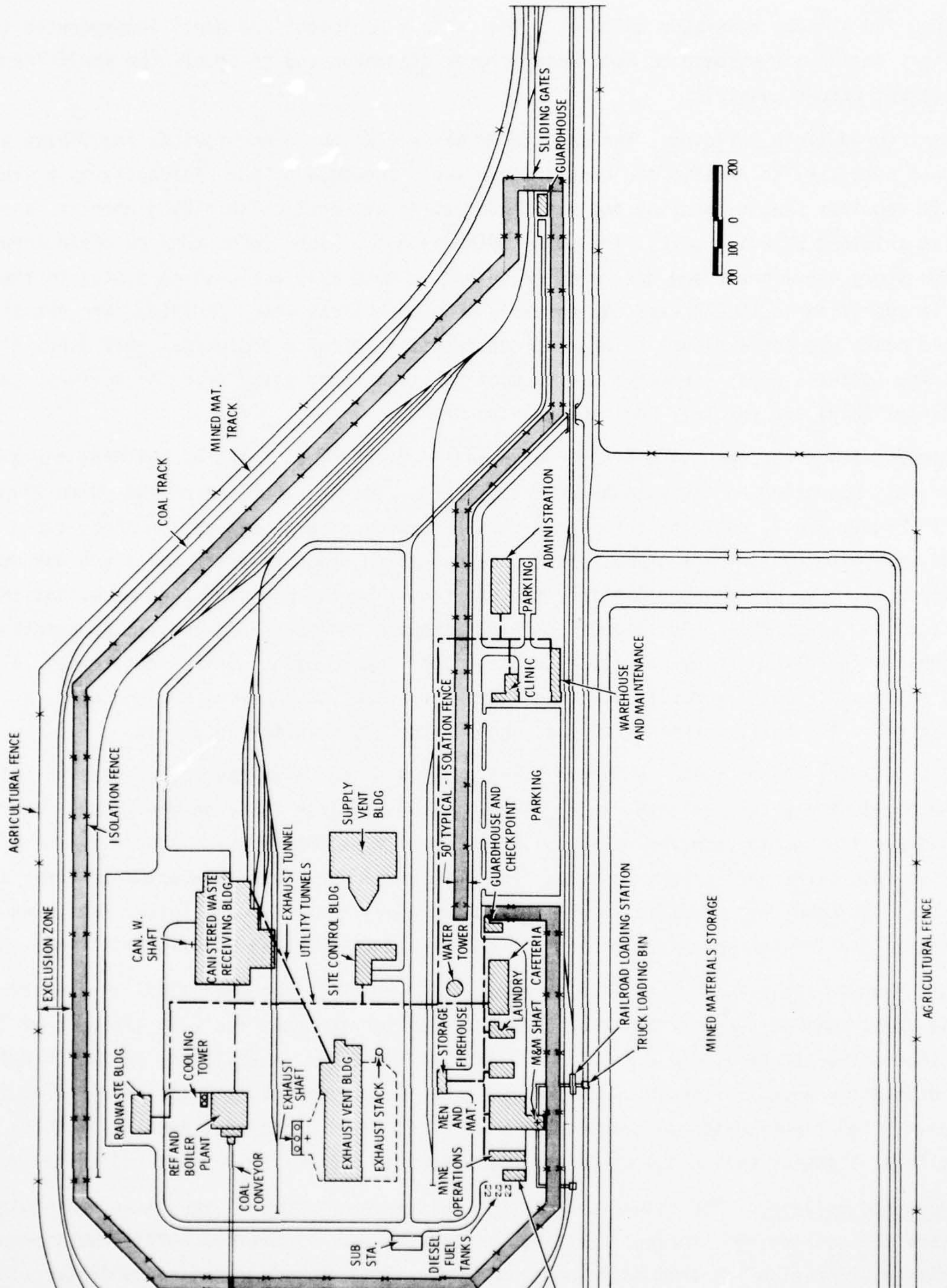


FIGURE 7.4.3. Facility Plot Plan - Once-Through Fuel Cycle Repositories

building, and standby generator building. The various equipment and parts incorporated into the repository design are assumed to meet manufacturer standards and to be off the shelf items to the greatest extent possible.

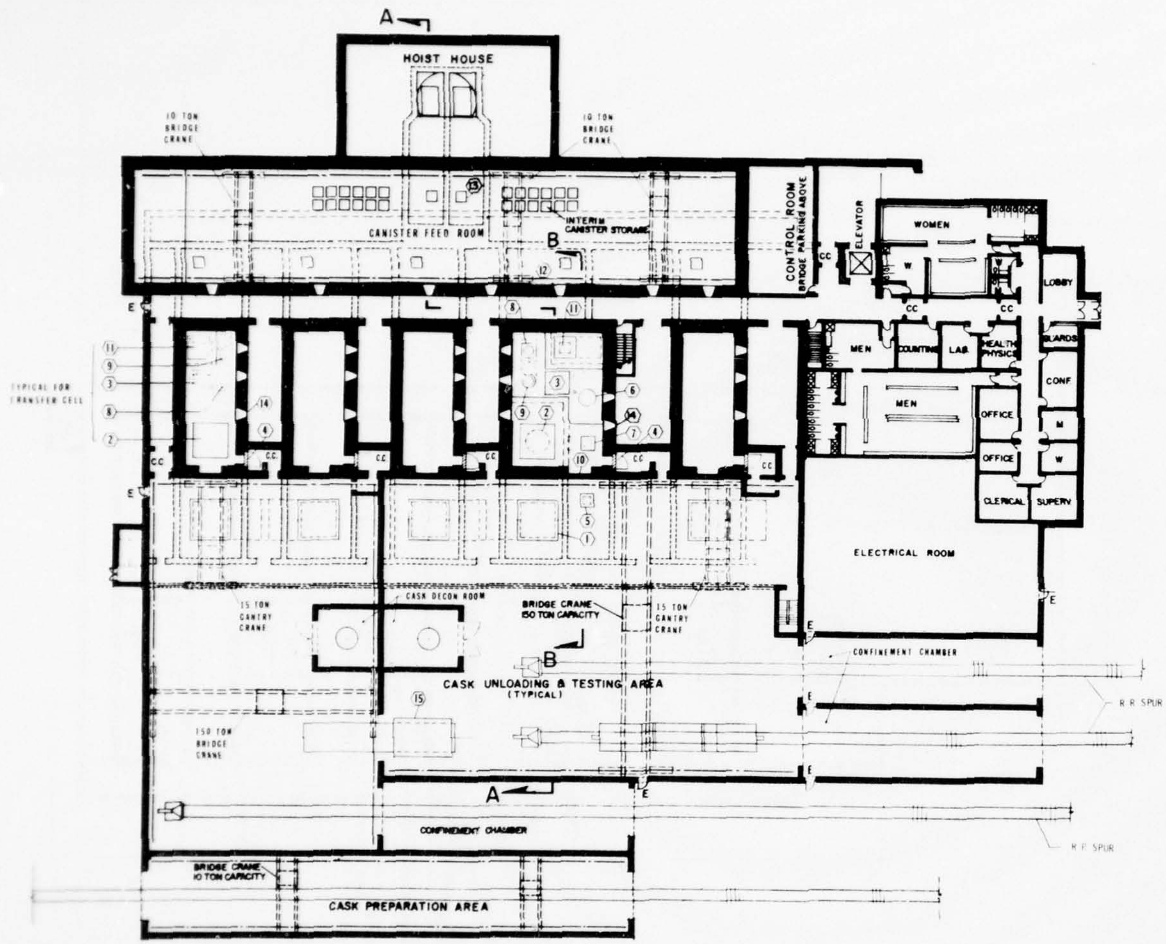
Canistered Waste Building. The canistered waste (CW) building provides facilities and equipment necessary to receive and handle spent fuel. Because of the radioactivity of the spent fuel, it requires remote handling and a high degree of attenuation shielding when it is removed from its shielded shipping cask. The CW building is a two-story reinforced concrete structure with one story above grade and one below grade and is generally designed according to the basic criteria specified in 10 CFR Part 50, Appendix A. Those areas where canisters are not inside shielded casks use conventional reinforced concrete for adequate biological shielding. All activities in these areas (transfer cells, canister feed room, waste transfer gallery, and shaft transfer gallery) are remotely monitored and controlled.

Figures 7.4.4 through 7.4.6 provide plan and section views of the CW building for a repository in salt operating in the oncethrough fuel cycle. The ground floor of the CW building, shown in Figure 7.4.4, contains a railroad car washdown and cask preparation shed, two independent bays with confinement chambers for cask unloading, four transfer cells and one overpack transfer cell, five operating galleries, the upper part of the canister feed room, and the electrical equipment room. Additional space for support functions and activities associated with canister handling include a lab, a counting room for counting surface dose rates, a health physics office, a security personnel check, a reception station, change rooms, a briefing room, the shift foreman's office, and service and maintenance areas.

The basement floor, shown in Figure 7.4.5 and 7.4.6, contains four cask transfer galleries and one overpack transfer gallery, which work with the transfer cells on the ground floor. Also included are five waste transfer galleries, five service galleries associated with the transfer galleries, the lower part of the canister feed room and an adjoining operating gallery, a mechanical equipment room containing air intake and exhaust filters and intake fans, and a maintenance and service area. Adjoining the canistered waste shaft are two shaft feed galleries.

As shown in Table 7.4.1 repositories located in the other geologic media are required to receive spent fuel at higher rates than are indicated for the salt facility (because of the longer operating periods). As a result, the canister handling capabilities of the CW building are modified for repositories in the other media by the additon of canister handling equipment and cells. The repositories in granite and basalt require 3 additional handling modules (transfer gallery, transfer cell etc.) while the repository in shale requires one additional module.

Radwaste Building. The radwaste treatment building contains all the tanks and equipment necessary for collecting, storing, treating, and solidifying liquid radioactive waste generated on the site. The plan for this building provides separate rooms for the concentrator, ion exchange systems, solidification systems, and truck access. Trucks bring in empty drums and chemicals and carry palletized filled drums to the CW building where they are lowered into the mine. The radwaste building also contains a guarded check-in of personnel, a change room, wash rooms, and a supervision room.



TYPICAL ARRANGEMENT

1 CASK LOWERING PORT
2 CASK UNLOADING PORT
3 CANISTER SURGE STORAGE
4 PERSONNEL ACCESS DOOR
5 ENTRY PORT-OVERPACK TRANSFER GALLERY
6 OVERPACK LID WELDER
7 EXIT PORT-OVERPACK TRANSFER GALLERY
8 CANISTER DECON WELL
9 CANISTER LEAK TEST WELL
10 EQUIPMENT TRANSFER DOOR TO CELL
11 ENTRY PORT-WASTE TRANSFER GALLERY
12 EXIT PORT-WASTE TRANSFER GALLERY
13 ENTRY PORT-SHAFT FEED GALLERY
14 VIEWING WINDOW
15 CASK TRANSFER CAR
E = EMERGENCY DOOR
CC = CONFINEMENT CHAMBER

FIGURE 7.4.4. Canistered Waste Building Ground Floor Plan, Once-Through Fuel Cycle, Repository in Salt

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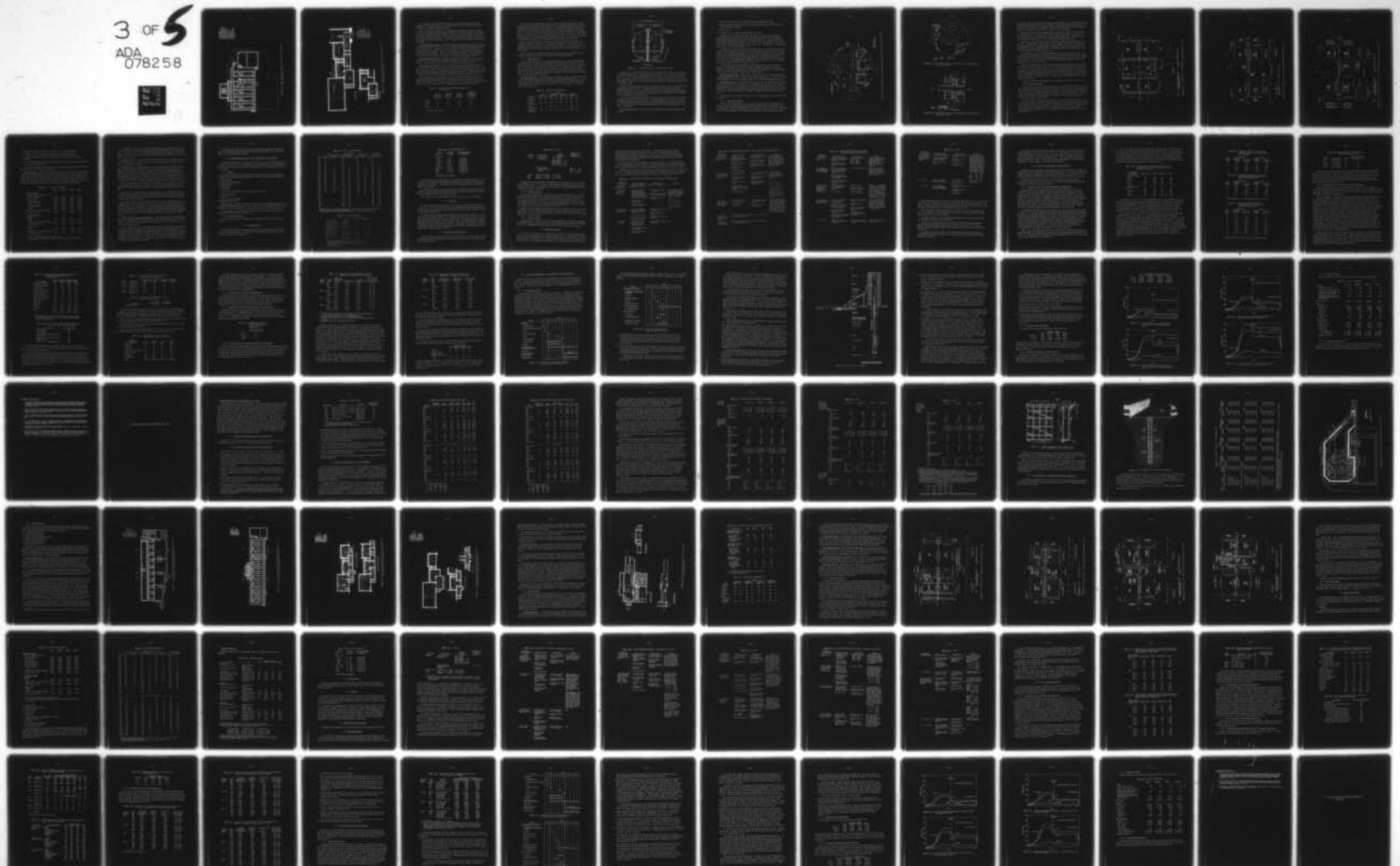


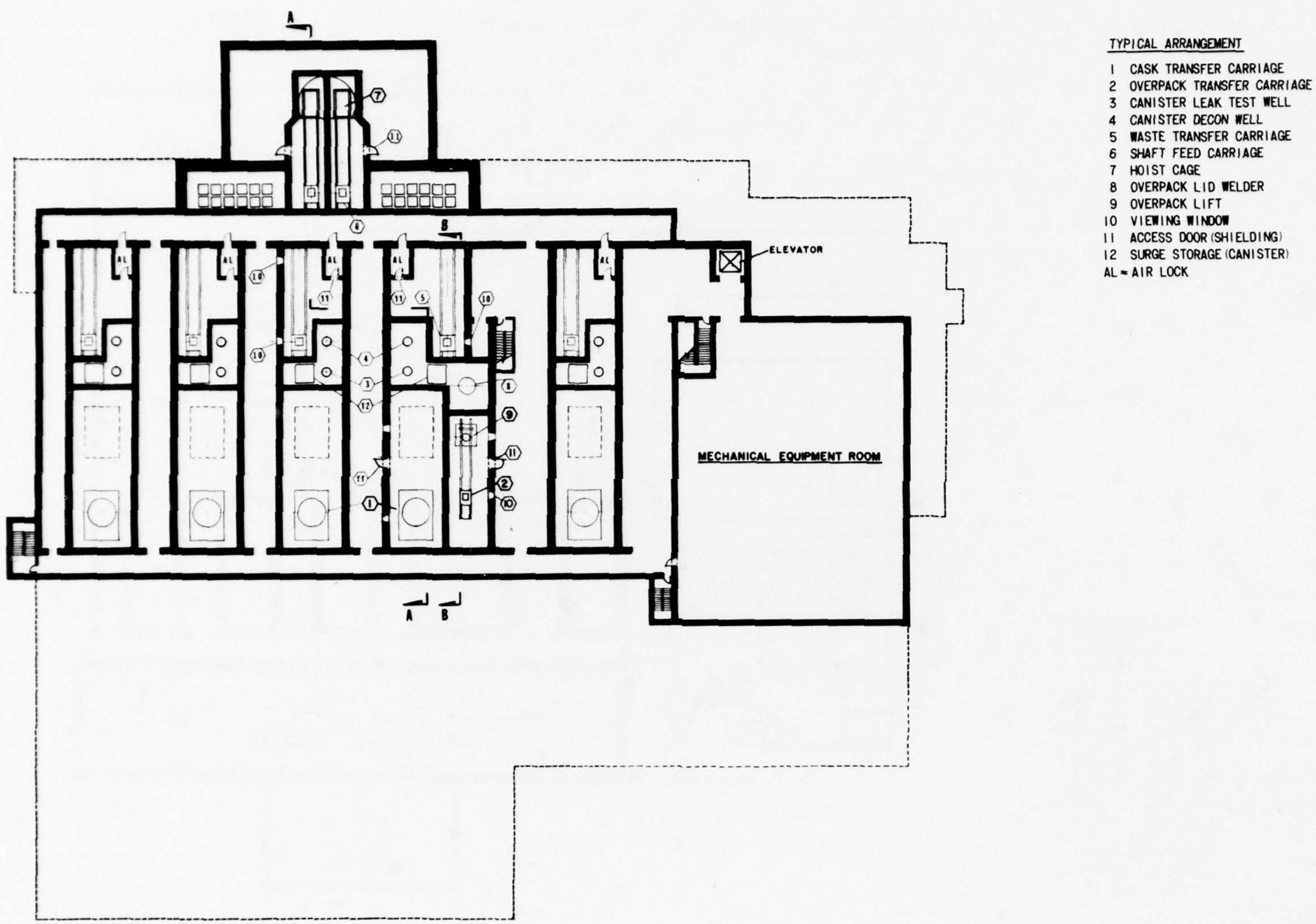


FIGURE 7.4.5. Canistered Waste Building Basement Plan, Once-Through Fuel Cycle, Repository in Salt

FIGURE 7.4.6. Canistered Waste Building Section Views, Once-Through Fuel Cycle, Repository in Salt

Liquid radwaste is generated in three major areas: the canistered waste receiving building, the solid waste incineration and compacting facilities in the exhaust ventilation building, and the radwaste treatment building.

Waste for treatment is pumped into the concentrator system where its volume is reduced. The vapors are condensed, and the condensate cooled and, if required, treated in ion exchangers. The residue from the concentrator is pumped to a holding tank for cement solidification. The ion exchange effluent is monitored in catch tanks and either pumped to the clean water tanks or recirculated through the concentrator system. The spent resin is also treated by immobilization with cement.

Exhaust Ventilation Building. The exhaust ventilation building houses the exhaust fans and filters that continuously filter the exhaust air from the canistered waste placement operations in the storage areas. In addition, the building contains the exhaust fans associated with mining and drilling operations. Space is also provided for incineration and/or compaction of site-generated solid radwaste. Exhaust air from placement operations is discharged through a 110 m (300 ft) high stack while mining exhaust air is discharged through an evasé stack.

Supply Ventilation Building. The supply ventilation building houses the fans that supply ambient air for the mining and waste emplacement operations. Heating coils for possible pre-heating of the mine supply air and filters for the intake air are also housed here.

Men and Materials Building. The men and materials building provides separate personnel facilities such as locker rooms and showers for the mining crews and the placing crews. It also has briefing rooms, check-in, and control facilities and would provide covered access to the men and material shaft headhouse for personnel, equipment, and materials. Excavated backfill rock is routed through this building to the men and materials shaft for transport to the mine.

Mined Rock Storage. During excavation of repository subsurface areas, all mined rock is brought to the surface and stored onsite. A portion of the rock is returned to the mine as backfill after completion of readily retrievable operations. Rock not used for backfill remains piled on the surface. Quantities of rock removed and stored are described in Table 7.4.4.

TABLE 7.4.4. Mining and Rock Handling Requirements at the First Repository

	Mined Quantity ($MT \times 10^6$)	Room Only Backfill ($MT \times 10^6$)	Total* Backfill ($MT \times 10^6$)	Permanent Onsite Surface Storage ($MT \times 10^6$)
Salt	30	14	17	13
Granite	77	29	38	39
Shale	35	15	21	14
Basalt	90	32	46	44

* Including room backfill.

Standard earth moving equipment is used to construct the storage area and pile. After grading of the stockpile areas at the repository in salt, and before stockpiling mined salt, an impermeable lining of hypalon covered by 0.6 m (2 ft) of montmorillonite type clay is placed over the entire stockpile area. The hypalon and clay function as a ground water protection barrier. As the pile is constructed, an asphalt cover is spread over the salt. All around the stockpile a trench with the same type of protection is constructed to collect any runoff water and transport it for the required treatment. Depending on specific site conditions, an evaporation pond or water treatment plant may be used to treat the water that is collected.

Some shales contain significant quantitites of soluble minerals that could leach into streams and groundwater. Moreover in a cold climate, freezing of the wet rock will result in fragmentation and liberation of clay particles that could result in local water pollution. The shale storage area is covered with a blanket of montmorillonite clay and surrounded by a collecting ditch.

Granite and basalt are rocks which generally do not contain noxious substances and are insoluble. Therefore, the storage pile area does not need special treatment.

7.4.4.2 Shafts and Hoists

The conceptual repositories for the once-through fuel cycle in salt and shale formations require three shafts to support waste handling and mining operations. These shafts are the canistered waste (CW) shaft, the men and materials (M&M) shaft, and the ventilation exhaust (VE) shaft. In addition to these three shafts, the repositories in granite and basalt require a mine production (MP) shaft. These shafts all differ in size, design, use and functional constraints. Table 7.4.5 summarizes shaft depths and diameters for the once-through cycle repositories in salt, granite, shale and basalt.

Canistered Waste Shaft. The canistered waste (CW) shaft is used to transport the canisters of spent fuel from the canistered waste building to the subsurface emplacement areas. The top of the shaft forms an integral part of the canistered waste building. The CW shaft has an inside diameter of 4.3 m (14 ft) and is lined with 0.3 m (1 ft) of concrete. The CW shaft is illustrated in Figure 7.4.7.

TABLE 7.4.5. Shaft Depth and Diameters, m

	Medium							
	Salt		Granite		Shale		Basalt	
Shafts	Depth	dia	depth	dia	depth	dia	depth	dia
Canistered waste	580	4.3	620	4.3	460	4.3	600	4.3
Men and materials	580	9.4	670	8.8	510	9.4	640	9.2
Mine production	---	---	630	8.8	---	---	600	9.2
Ventilation exhaust	560	7.9	610	8.5	440	7.9	580	8.5

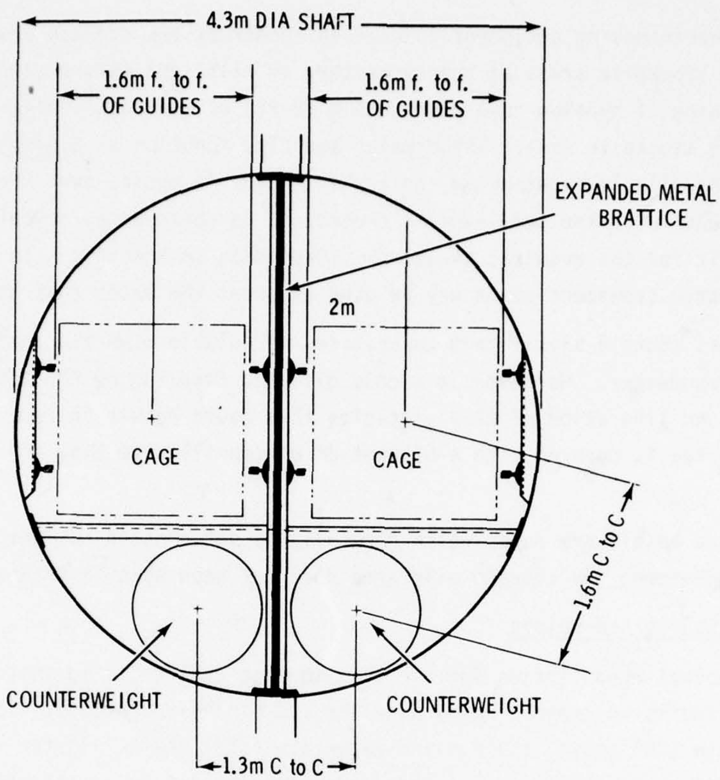


FIGURE 7.4.7. Canistered Waste Shaft - Once-Through Fuel Cycle

The CW Shaft is divided into two sections, with a separate hoist in each. Four canisters are lowered by each hoist in a specially designed cage with 5.1 cm (2 in.) thick stainless steel walls, a 12.7 cm (5 in.) thick lead core and a 0.64 cm (0.25 in.) steel plate liner retainer skin. The cage has a weight of 91 MT (100 tons) and is hoisted in balance with a 91 MT (100 tons) counterweight. Reliability of canister delivery to the repository is provided by the dual cage counterweight systems.

Because of the heavy loads being hoisted, friction type hoists are used. These hoists are tower-mounted over the CW shaft. To ensure safe operation of this facility, the shaft is equipped with buntons at 1.2 m (4 ft) centers, and the cages are guided by elevator-type guide rails. The cages are also equipped with an elevator-type braking mechanism which is designed to stop the cage during an overspeed condition.

In the mine, the shaft provides access to the BWR receiving station at one elevation and to the PWR receiving station 23 m (75 ft) lower. The PWR and BWR receiving stations are located on different elevations to provide space for routing men-and-materials corridors to each mine area. The receiving stations also provide shielded facilities for the remote transfer of canisters from the shaft.

Men and Materials Shaft. At the salt and shale repositories the men and materials shaft provides:

- access for men and materials into and out of the waste emplacement areas
- means for removing the mined rock during construction and returning backfill material during emplacement operations
- utility access
- a supply duct for introducing ventilation air to the mine.

The M&M shaft is divided into four main compartments: personnel and equipment cage, excavated material hoisting skips, auxillary cage and utility access space. In addition, approximately 42,000 m^3/min (1,500,000 ft^3/min) of ventilation air are supplied through this shaft over the entire shaft cross section. The shaft is lined with at least 0.3 m (1 ft) of concrete with finished inside diameters listed in Table 7.4.5. The M&M shaft at a repository in salt or shale is illustrated in Figure 7.4.8.

The M&M shafts in granite and basalt are divided into two main compartments that provide access for men and materials into and out of the waste emplacement areas, utility access, and a supply duct for introducing ventilation air into the mine. Utility access to the repository is via the men and material compartment and approximately 42,000 m^3/min (1,500,000 cfm) of ventilation air is supplied to the mine through this shaft. The M&M shaft at a granite repository is 8.8 m (29 ft) in diameter while the M&M shaft in basalt is 9.1 m (30 ft) in diameter. This shaft in granite and basalt is shown in Figure 7.4.9.

The personnel and equipment compartment contains a double-deck cage with capacity for 140 men. The compartment is also large enough to accommodate large pieces of equipment in assembled or partially disassembled form. The personnel and equipment hoist is a tower mounted friction type, which operates in balance with a counter weight.

Ventilation Exhaust Shaft. Ventilation air from the mine areas is exhausted through the ventilation exhaust shaft into filtration systems located on the surface. The shaft is divided into two compartments to provide separate exhaust for the mining and emplacement operations.

Mine Production Shaft. Substantially larger amounts of rock are excavated from granite and basalt than from salt or shale repositories. To accomodate the higher rate of rock removal, a mine production (MP) shaft is provided at the granite and basalt repositories (Figure 7.4.10). The MP shaft contains skip hoist equipment for removal of mined rock to the surface and supplies an additional 31,000 to 37,000 m^3/min (1,100,000 to 1,300,000 cfm) of ventilation air to the mine.

Shaft dimensions are summarized in Table 7.4.5 for the repositories in salt, granite, shale and basalt.

7.4.4.3 Subsurface Facilities

The repository underground layouts are conventional room and pillar arrangements that provide for repository ventilation, opening stability, thermal effects, and efficient use of excavated space. The overall underground area is bounded by an upper limit of 800 ha (2000 acre), which is based on reasonable waste storage capacity and efficient waste transport distances.

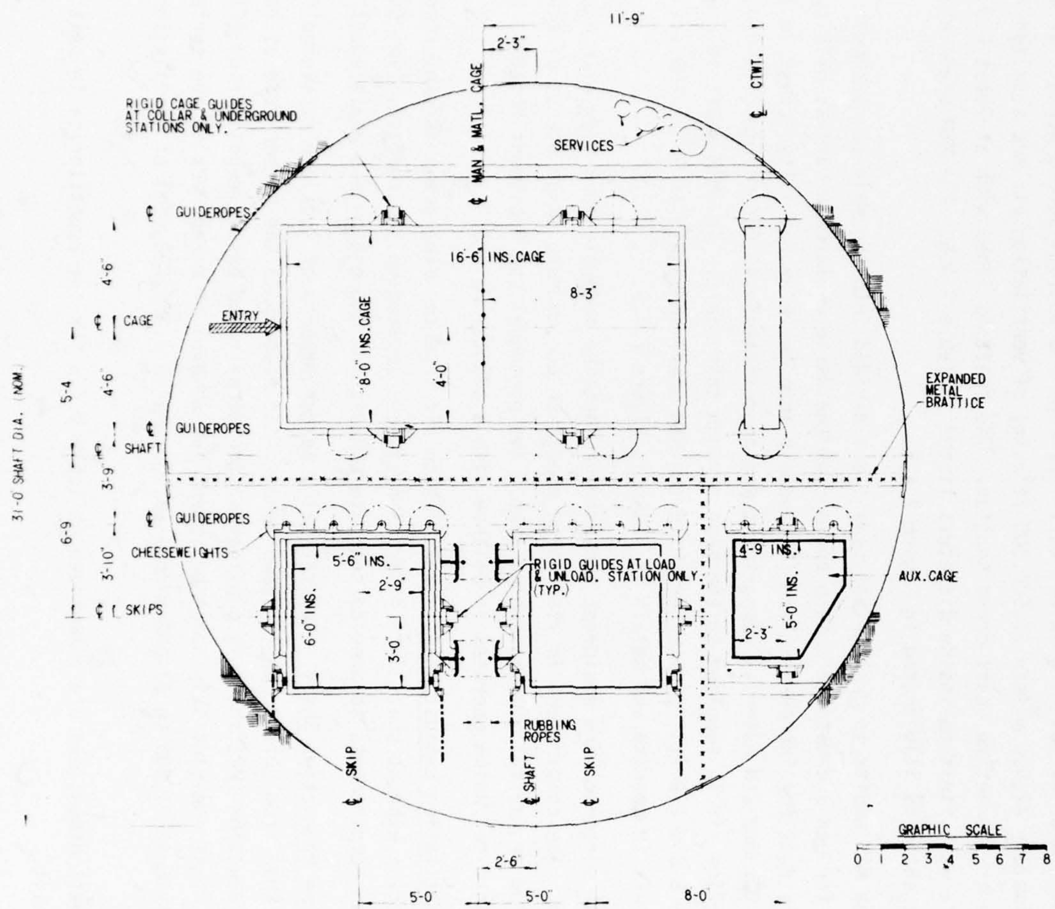
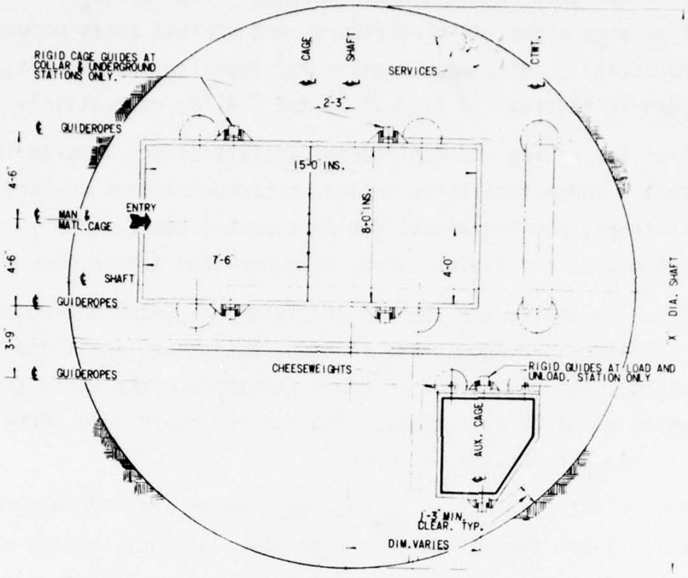


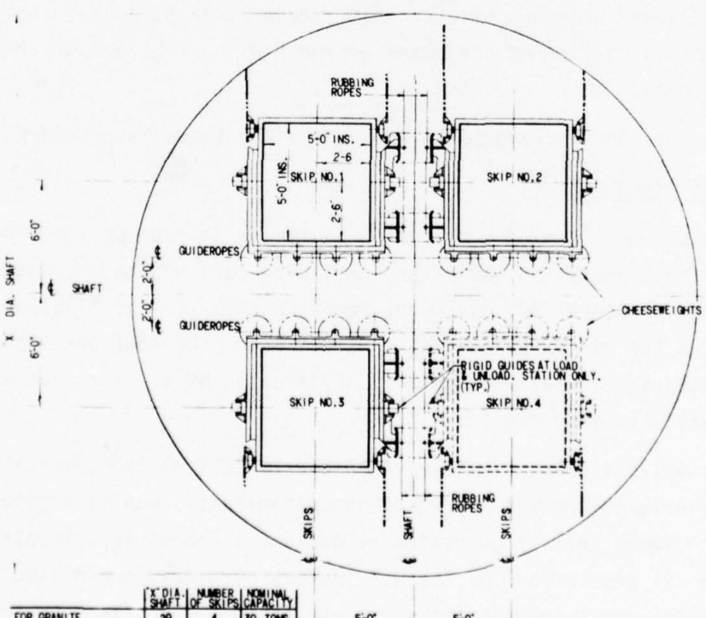
FIGURE 7.4.8. Men and Material Shaft - Once-Through Fuel Cycle Repositories in Salt or Shale



	X" DIA. SHAFT
FOR GRANITE	29
FOR BASALT	30

PLAN OF MEN AND MATERIALS SHAFT

FIGURE 7.4.9. Men and Material Shaft - Once-Through Fuel Cycle Repostories in Granite and Basalt



	X" DIA. SHAFT	NUMBER OF SKIPS	NOMINAL CAPACITY
FOR GRANITE	29	4	30 TONS
FOR BASALT	30	4	30 TONS

5-0" 5-0"

PLAN OF PRODUCTION SHAFT

FIGURE 7.4.10. Mine Production Shaft - Once-Through Fuel Cycle Repositories in Granite and Basalt

Of this 800 ha total area, spent fuel emplacement areas occupy 650 to 730 ha (1600 to 1800 acres), with shafts, general service areas, main corridors, and unmined areas occupying the remaining 80 to 160 ha (200 to 400 acres). Mine master plans for repositories in salt, shale, and granite and basalt are provided in Figures 7.4.11, 7.4.12 and 7.4.13, respectively.

Support facilities for mining and emplacement activities are located in the central shaft area at each repository. These facilities include a communications center, a warehouse and mechanical/electrical shops, parking areas, and equipment assembly areas. These facilities are located outside the 46 m (150 ft) radius safety zone provided around each shaft.

Extending out from the shafts are the men and materials (M&M) and waste transport main corridors. The M&M corridors provide access for men, equipment, and mined rock from the shafts to the emplacement areas. The M&M corridors are separate from the waste transport corridors, which are used for waste handling operations. The M&M corridors also carry ventilation supply air from the shafts to the appropriate mine areas.

At the repository in salt, spent fuel emplacement rooms are 1070 m (3500 ft) long. This long room arrangement provides for efficient use of the continuous mining machines that excavate this repository. The emplacement rooms are excavated directly off the main corridors in panels of four rooms with a pillar of intact salt 14 m (45 ft) wide between each room. Panels are separated by pillars 24 m (80 ft) wide.

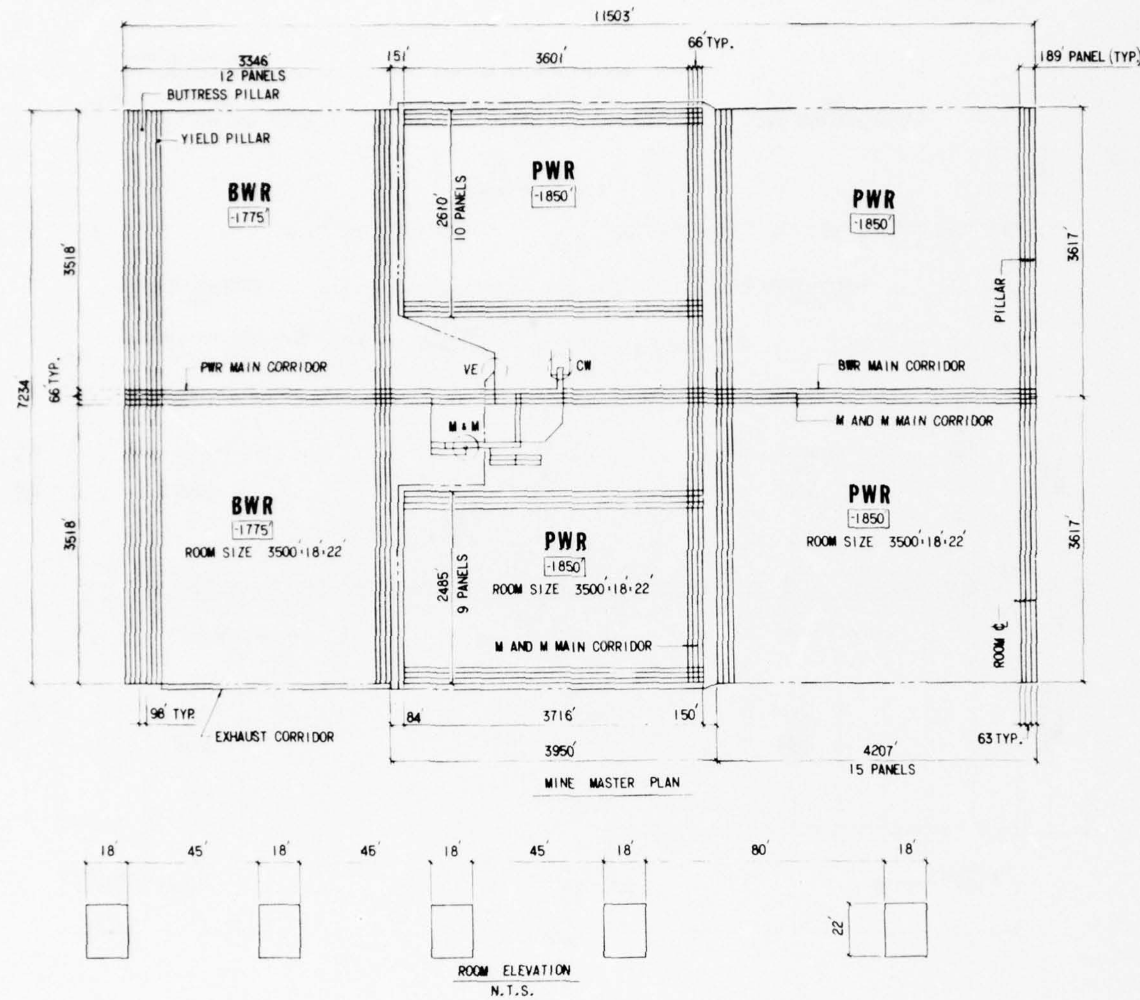
Because nonsalt repositories are excavated with conventional drill and blast techniques a short room layout (170 m (560 ft) long) is used in order to provide multiple working faces for mining crews. In this way, rubble removal, blasting, charging, and drilling activities can be carried on simultaneously by different crews working in different room panels. The emplacement rooms are excavated along branch corridors that extend at right angles from the main corridors. Each branch corridor is considered one panel of rooms with each room separated from the next by a pillar of intact rock.

A summary of repository arrangements and area allocations is provided in Table 7.4.2.

7.4.4.4 Ventilation Systems

The basic purpose of the mine ventilation system is to protect the mine personnel and the public from excessive exposure to radiation and mining dust and to provide an adequate supply of clean, fresh air for personnel and equipment requirements. The ventilation system is arranged so that the airflows for mining activities and for waste placement are separated. This ensures protection from excessive radiation exposure. Filtration of all air that may have come in contact with the waste is also provided.

An adequate supply of fresh air is also needed to control the chemical and physical quality of the mine atmospheric environment by removing contaminants such as engine exhaust gases, dust, heat, moisture, and smoke that is generated by equipment and mining processes. Ventilation air flow complies with all regulations of the U.S. Bureau of Mines, the Mining Enforcement and Safety Administration, and the strictest state regulations. Ventilation requirements include the following:



NOTES: 1. ALL LINES REPRESENT CENTER LINES OF ROOMS OR CORRIDORS
2. ALL CORRIDORS ARE 30' WIDE X 20' HIGH

FIGURE 7.4.11. Mine Master Plan - Once-Through Fuel Cycle Repository in Salt

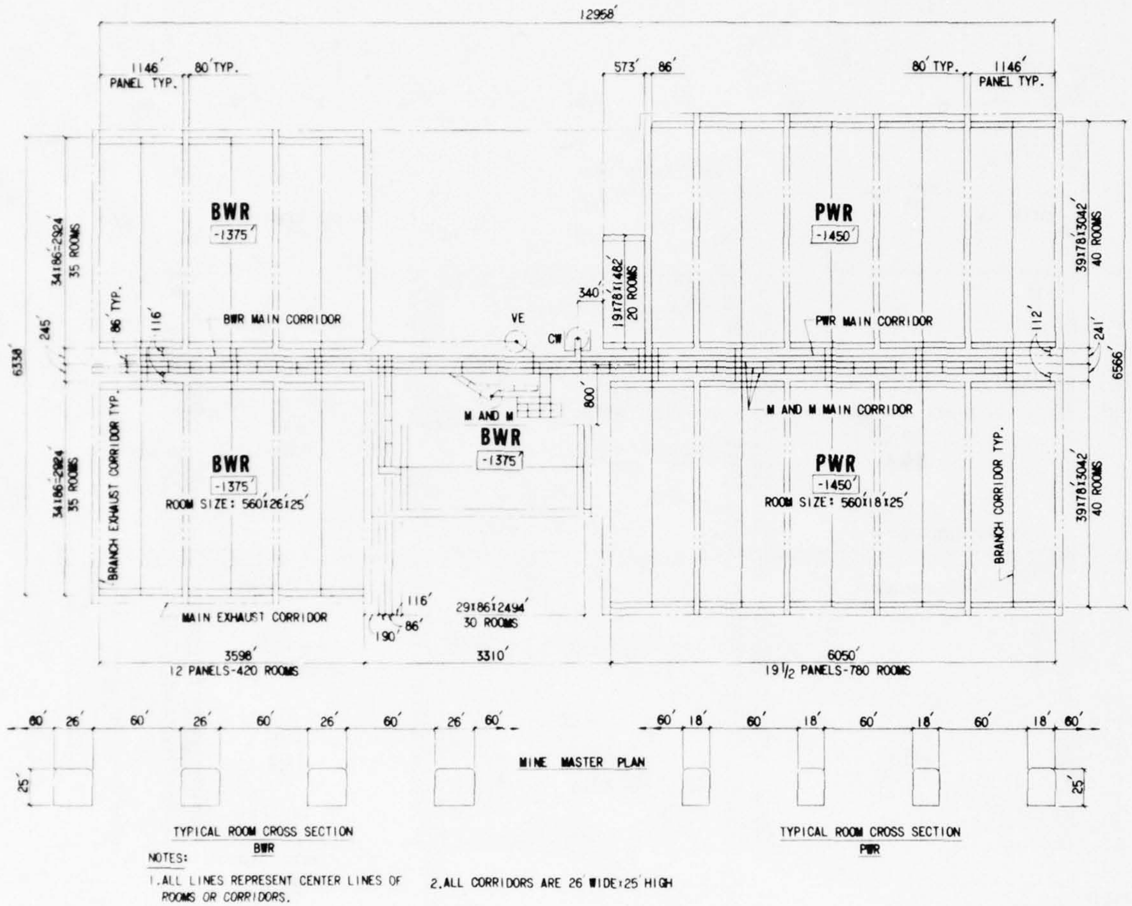
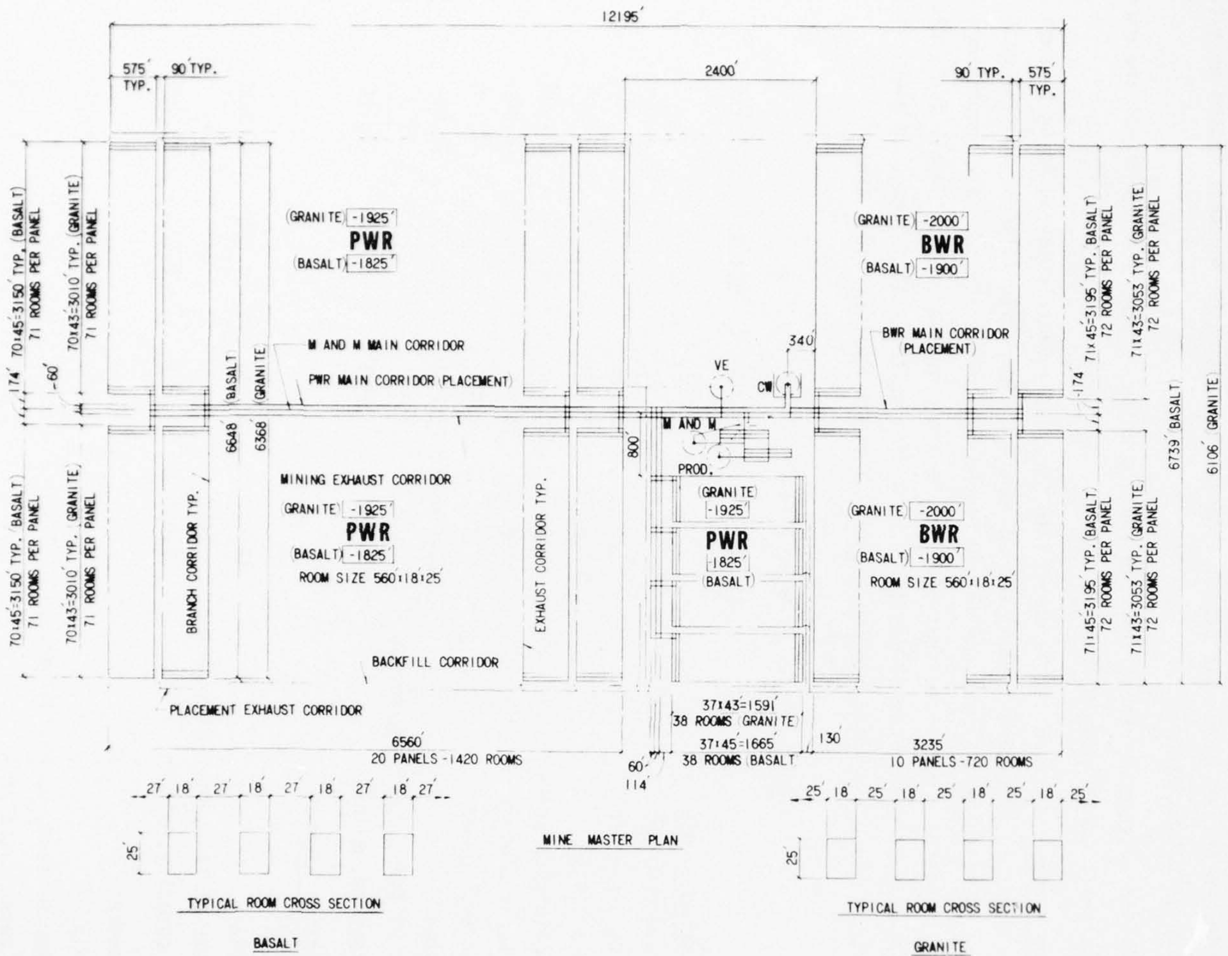


FIGURE 7.4.12. Mine Master Plan - Once-Through Fuel Cycle Repository in Shale



NOTES: 1. ALL LINES REPRESENT CENTER LINES OF ROOMS OR CORRIDORS
2. ALL CORRIDORS ARE 30' WIDE X 20' HIGH

FIGURE 7.4.13. Mine Master Plan - Once-Through Fuel Cycle Repositories in Granite or Basalt

- minimum of 5.7 m^3/min (200 ft^3/min) of fresh air for each man underground
- minimum of 3.5 m^3 (125 ft^3)/brake horsepower of diesel-powered equipment
- minimum of 15 m^3/m^2 (50 ft^3/ft^2) of headings (or 15 linear m/min of air velocity in entries)
- 1,100 m^3 (40,000 ft^3) of fresh air per continuous mining machine for salt (based on existing practice in potash mines).

The ventilation systems at repositories in salt, granite, shale, and basalt for the once-through fuel cycle are summarized in Table 7.4.6.

Ventilation Supply. Outside air is drawn into the supply ventilation building through air intake louvres (equipped with bird screen) and 30% NBS efficiency air filters. During cold weather, heating coils located in the air intake plenum maintain the supply air temperature at a minimum of 4.4°C (40°F). Motorized shut-off dampers are provided in each supply fan's discharge. The sub-surface supply air is monitored by a flow switch that triggers an alarm if the air flow is reduced below a pre-set minimum because of fan failure or other causes.

TABLE 7.4.6. Mine Ventilation Summary

	Salt	Granite	Shale	Basalt
Ventilation Supply				
Fresh air required, m^3/min[(a)]				
Mining operations	11,000	25,000	11,000	27,000
Hole drilling	7,800	7,800	7,800	7,800
Miscellaneous areas	9,900	22,000	9,900	25,000
20% Recirculation	5,800	11,000	5,800	12,000
Waste Emplacement	7,100	7,100	7,100	7,100
Total	42,000	73,000	42,000	79,000
Number of fans, operating/backup	6/2	11/2	6/2	11/2
Fan capacity, m^3/min	6,800	7,100	6,800	7,100
Ventilation Exhaust				
Mining[(a)]				
m^3/min	34,000	68,000	34,000	68,000
Number of fans, operating/backup	2/1	4/1	2/1	4/1
Fan capacity, m^3/min	17,000	17,000	17,000	17,000
Emplacement[(b)]				
m^3/min	7,100	7,100	7,100	7,100
Number of fans, operating/backup	1/1	1/1	1/1	1/1
Fan capacity, m^3/min	7,100	7,100	7,100	7,100

a. Air is filtered through a 30% roughing filter.
b. Air is filtered through a 90% roughing filter followed by two HEPA filters in series.

Air discharged into the supply air plenum travels down the men and materials shaft, or in the case of the granite and basalt repositories, down both the M&M shaft and mine production shaft. In the mine, the airflow is divided into separate supply air systems by a controlling damper. A throttle in each of the main supply branches permits the ventilation air supply to be shut off in case of fire.

Proper distribution of the ventilation air to the working areas is accomplished through the use of stoppings, doors, air locks and crossings in the main airways, and line brattices and auxiliary fans at the working faces.

Ventilation Exhaust - Mining Operations. All corridors and rooms are mined by multiple passes of continuous mining machines in salt and by conventional drill and blast methods in granite, shale, and basalt. During all mining in non-salt media and during the first pass in salt mining, dust is effectively removed from the working face by locating the inlet tube of a portable ventilating exhaust fan as close to the face as possible. The fan exhausts the dusty air through reinforced flexible tubing into the main corridor which serves exclusively as a mining exhaust air corridor. Additional lengths of ducting are added as the mining advances.

During the second pass of the continuous mining machine in salt, an exhaust fan is placed in the adjacent, mined storage room. Exhaust air is drawn past the working face into the exhaust corridor that connects the rooms and is exhausted through the adjacent, mined storage room.

Hole drilling, trenching, and sleeve placement operations are ventilated the same way as the second pass mining operation in salt. Supply air would flow through the entire room where the work is in progress, and would be exhausted through the adjacent unoccupied storage room into the mining air exhaust corridor.

All exhaust air from the various mining operations and from the maintenance shop area is collected by the mining air exhaust corridor, which is separated by well-sealed stoppings from the M&M corridors, which supply fresh air. Where the exhaust air corridor crosses entryways used by men and equipment, the exhaust air passes above the M&M corridor through an aerodynamically shaped overcast. The exhaust air corridor is connected to the mining exhaust compartment of the ventilation exhaust shaft, through which the air travels to the ventilation exhaust building.

Ventilation Exhaust - Emplacement Operations. Areas where waste handling and emplacement operations are being conducted are maintained at a slightly lower pressure than the rest of the repository. Therefore, any cross leakage will be from clean areas in the repository to the potentially contaminated areas.

Initially spent fuel canisters are emplaced at the rear of the emplacement rooms (the end nearest the periphery ventilation duct). Emplacement proceeds toward the front of the rooms. Ventilation air supplied through the placement corridor flows through the emplacement room and is exhausted through the periphery ventilation duct system. As a result, emplacement operations are always conducted in a fresh air stream with the air flowing past the emplacement point. When emplacement is completed in a given room, the room is temporarily sealed and the ventilation airflow is directed into the next room.

The periphery ventilation duct carries exhaust air from waste handling and emplacement areas to the emplacement exhaust compartment of the ventilation exhaust shaft. The air then flows to the ventilation exhaust building.

7.4.5 Operating Requirements for the Once-Through Fuel Cycle Repository

The repository operates on a 5-day week, three shifts per day. The waste handling facilities allow reserve capacity for any irregularities in delivery schedule. Surge storage capacity is provided in both the above-ground and below-ground facilities. An operating efficiency of 67% is assumed.

7.4.5.1 Manpower

The operating labor force at the repositories includes a variety of craft, engineering, management, and miscellaneous support personnel involved in the following major activities:

- waste handling
- hole drilling and trenching
- sleeve placement
- back filling
- maintenance.

In support of these activities are various functions including:

- radiation monitoring
- utilities
- excess mined rock management
- general administration.

Operating labor requirements at the repositories are primarily dependent on waste receiving rate and will therefore vary from year to year. Table 7.4.7 lists estimated total labor force requirements by year of operation for the repositories in salt, granite, shale and basalt.

Labor requirements for construction of surface facilities and shafts, subsurface facilities, and mining are described in Section 7.4.11.

7.4.5.2 Supplies and Utilities

Repository supply and utility requirements are listed in Tables 7.4.8 and 7.4.9, respectively.

7.4.6 Secondary Wastes

Secondary wastes generated at the repository during spent fuel handling and storage are treated and disposed of onsite. Secondary wastes include decontamination wastes, air filter cartridges, and general contaminated trash.

TABLE 7.4.7. Repository Operating Staff

	Man Yrs/Yr							
	Salt		Granite		Shale		Basalt	
Year	Operations[a]	Drilling and Backfill[b]	Operations	Drilling and Backfill	Operations	Drilling and Backfill	Operations	Drilling and Backfill
1980	60		60		60		60	
1981	65		65		65		65	
1982	80		80		80		80	
1983	100		100		100		100	
1984	250		250		250		250	
1985	470	18	500	32	510	27	410	34
1986	510	45	540	85	550	67	460	90
1987	560	45	590	85	600	67	510	90
1988	550	45	580	85	590	67	500	90
1989	550	45	580	85	590	67	500	90
1990	600	190	620	310	630	230	550	410
1991	630	200	660	340	670	250	600	440
1992	630	200	660	340	670	250	600	440
1993	640	200	670	340	680	250	600	440
1994	640	200	670	340	680	250	600	440
1995	650	200	680	350	690	250	620	450
1996	670	210	700	380	710	270	640	490
1997	690	210	720	380	730	270	670	490
1998	710	210	750	380	760	270	700	490
1999	730	210	760	380	770	270	720	490
2000	760	220	800	430	810	300	760	540
2001			830	480	830	320	790	590
2002			860	480	860	320	830	590
2003			890	480			870	590
2004			920	480			910	590
2005			950	530			940	640
2006			970	570			980	690
2007			1000	570			1000	690
2008			1000	570			1100	690
2009			1000	570			1100	690

a. Operations includes: surface and subsurface waste handling and emplacement, general operations support (maintenance, utilities, etc.) and administrative personnel.
b. Drilling and backfill includes: hole drilling, trench excavation, sleeve emplacement and backfilling.

TABLE 7.4.8. Supply Requirements

		Average Annual Requirements			
Description	Use	Salt	Granite	Shale	Basalt
PWR Canister Overpacks[a] 41 cm dia x 5 m long	Contains damaged PWR canisters	5	7	5	7
BWR Canister Overpacks[a] 32 cm dia x 5 m long	Contains damaged BWR canisters	7	10	8	10
PWR Retrievability Sleeves[b] 46 cm dia x 7.2 m long	Emplacement Hole liners	2020	2020	2020	2020
BWR Emplacement and Retrievability sleeves 36 cm dia x 7.2 m long	Emplacement Hole liners	3520[b]	9850[c]	3520[b]	9850[c]
PWR Concrete Plug[b] 41 cm dia x 2.5 m long	Emplacement Hole Plug	2020	2020	2020	2020
BWR Concrete Plug 31 cm dia x 2.5 m long	Emplacement Hole Plug	3520[b]	3520[b]	3520[b]	3520[b]
BWR Trench rack[c]	Sleeve support during trenching	--	1230[d]	--	1230[d]

a. Overpack requirements are based on 0.1 of canisters received leaking or damaged.
b. Sleeves and plugs needed for first 5 years only.
c. BWR canisters are emplaced in trenches that require continual use of sleeves and racks. Therefore, annual quantities of BWR sleeves and racks in granite and basalt are for all years of operation.
d. Each rack holds 8 BWR canisters.

TABLE 7.4.9. Utility Requirements[a]

Utility	Use Rate	Annual Requirement
Coal	8.2 MT/hr	72,000 MT/yr
Electricity		
Salt	11 MW	96,000 MWh
Granite	15 MW	130,000 MWh
Shale	11 MW	96,000 MWh
Basalt	15 MW	130,000 MWh
Diesel fuel	1.5 m^3/hr	13,000 m^3/yr
Steam	91 MT/hr	800,000 MT/yr
Water	2 kg/hr	10,000 kg/yr

a. Peak yearly requirements.

Liquid waste is collected and piped to the radioactive waste treatment building. In this building the liquid wastes are concentrated by evaporation; the condensate receives further treatment in ion exchangers. The residual material from the concentrator is solidified with cement in 55-gal drums.

Contaminated solid wastes are collected and sent to the exhaust ventilation building for treatment. Combustible wastes are incinerated and the resulting ash immobilized with cement in drums; noncombustible wastes are compacted into drums.

After treatment, all secondary wastes are lowered into the repository for disposal.

7.4.7 Emissions

All wastes arriving at the repository are fully contained in leak tested steel canisters or steel drums. As a result, the only sources for airborne emissions from these waste containers are handling accidents that could damage and breach the canisters. Potential accidents are described in Section 7.4.9.

An estimate of the integrated annual release due to minor accidents (Section 7.4.9) for this facility is included in Table 7.4.10 in addition to annual releases from onsite coal and diesel combustion and the ventilation system cooling tower. The integrated annual release was developed by weighting the minor accident releases by their expected frequencies and summing the quantities for all identified minor accidents. In addition, a contingency was included in the integrated release to account for unidentified minor accidents and to compensate for the uncertainty in expected frequency information.

7.4.8 Decommissioning Considerations

The waste repository surface facilities have been designed for a 30-year useful lifetime. Upon completion of waste emplacement operations, the repository facilities will require some form of decommissioning.

TABLE 7.4.10. Emissions

Emission	Description	Annual Quantity	Radioactivity Release to Atmosphere, Ci
Gaseous	Coal and diesel combination	SO_x - 610 MT CO - 150 MT Hydrocarbons - 54 MT NO_x - 940 MT Particulates - 27 MT Heat - 2.4 x 10^7 MJ	
	Minor accident integrated annual release		^{85}Kr - 5 ^{14}C - 1 x 10^{-4} ^{129}I - 1 x 10^{-5} ^{3}H - 1 x 10^{-2}
Cooling tower water	Evaporated (T=39°C) Drift (T=38°C) Blowdown (T=27°C)	9.2 x 10^7 kg 4.4 x 10^5 kg 1.6 x 10^7 kg	

Technology is currently available for decommissioning nuclear facilities (e.g., reactors, fuel reprocessing plants, etc.) and can be readily modified to decommission facilities at a geologic isolation site. These procedures, which encompass alternative decommissioning modes, are discussed in detail in Section 8.

7.4.8.1 Subsurface Facility

Initially, the repository will be operated in a readily retrievable mode for a period sufficient to perform certain tests related to acceptability of the geologic formation and site. If this test period does not confirm the earlier tentative conclusion that the site is acceptable, the wastes will be removed and the facility decommissioned. Removal of high-level radioactive materials from a temporary experimental installation in a mine was successfully demonstrated on a small scale in Project Salt Vault.[(4)]

Repositories found acceptable during the test phase will continue to receive waste. When the repository is full, the emplacement operations will cease and the facility will be decommissioned. This will involve the backfilling of all mined volumes and the filling, plugging and sealing of all access shafts.

7.4.8.2 Surface Facilities

Repository surface facilities will also require decommissioning. Regardless of decommissioning mode (dismantling, safe storage, etc.), the procedures are well defined for these types of facilities. Decommissioning wastes may be shipped offsite or placed in the repository.

7.4.9 Postulated Accidents

All structures are maintained at a negative pressure relative to the atmosphere, and all entries into and from confinement areas are made through air locks. Contamination is controlled by directing air flow from areas of less contamination potential to areas of increasing contamination potential. Air discharged from confinement areas is exhausted through a prefilter and

two high-efficiency particulate air (HEPA) filters. Ventilation systems are backed up by standby facilities to maintain confinement in the event of fan breakdown, filter failure, or normal power outage. Automatic monitoring of all potential sources of contaminated effluents is provided with remote readout and alarm at both the central control room in the mine operations building and the guardhouse.

Repository site selection considerations, discussed in Section 7.2.6, are general geologic criteria for acceptable repository locations. These factors combine to provide maximum assurance that the repository integrity will be maintained throughout the hazardous lifetime of the waste.

Postulated minor and moderate accident scenarios for the repositories are given in Tables 7.4.11 and 7.4.12. There were no severe (as defined in Section 3.7) accidents identified for the repository. Several extremely improbable non-design basis accidents are described in Table 7.4.13.

TABLE 7.4.11. Postulated Minor Accidents - Once Through fuel Cycle Repositories

Accident No. and Description	Sequence of Events	Safety System	Release
7.1 — LLW drum rupture due to handling error.	(TRU LLW not produced in this fuel cycle. This accident is described in Section 7.5.)		
7.2 — Minor canister failure. Estimated frequency $\sim$1/yr	1. Rough handling during transportation and unloading or presence of canister defect results in the formation of a pin hole leak in canister containing a failed fuel rod. 2. Leak detected in receiving facility. 3. Canister overpacked and placed in storage.	1. Canisters inspected prior to shipment. 2. Canisters pressurized with helium for leak detection. 3. Overpacking facilities available	One failed pin in a PWR fuel assembly releases gaseous activity from the canister. The following activity is released over a 2 day period. ^{85}Kr - 3 Ci ^{14}C - 4×10^{-5} Ci ^{129}I - 5×10^{-6} Ci ^{3}H - 5×10^{-3} Ci Others - negligible
7.3 — Receipt of externally contaminated canister.	1. Canister received with smearable contamination above specification limits. 2. Contamination detected. 3. Canister decontaminated and placed in storage.	1. Canisters inspected prior to shipment. 2. Canister inspected on receipt. 3. Decontamination facilities available.	None
7.4 — Dropped shipping cask.	1. Equipment failure or operator error drops shipping cask into transfer gallery. 2. Cask inspected. Expected to be undamaged. 3. Impact absorber removed and replaced.	1. Impact absorber minimizes cask damage.	None

TABLE 7.4.12. Postulated Moderate Accidents - Once Through Fuel Cycle Repositories

Accident No. and Description	Sequence of Events	Safety System	Release
7.5 – Waste Container drop during handling.	1. Equipment failure or operator error results drop sufficient to fail waste container. 2. Particulate release to cell filters. 3. Container contents repackaged. 4. Area decontaminated.	1. Drop height minimized by facility design. 2. Cell HEPA filters reduce release to atmosphere. 3. Repackaging and decontamination facilities available.	The maximum drop height for any waste form is 20 m. This is much less than a drop down the mine shaft. Therefore it is expected that releases from this accident would be much less than Accident 7.6.
7.6 – Waste package dropped down mine shaft. Estimated frequency 1×10^{-5} per year.	1. Canistered waste shaft hoist fails. 2. Hoist cage containing 4 canisters drops to mine level. 3. Canister is breached on impact. 4. Canister contents repackaged using specially designed remotely operated equipment. 5. Area decontaminated.	1. Failsafe wedge type braking system on hoist cage. 2. Mine exhaust filter system reduces atmospheric releases. 3. Repackaging and decontamination equipment available.	Four PWR assemblies (2 MTHM) are dropped releasing the following activities over a one hour period. ^{3}H - 6 Ci ^{14}C - 4×10^{-2} Ci ^{85}Kr - 4×10^{3} Ci ^{129}I - 6×10^{-3} Ci Other FPs - 1% of Table 3.3.8 per MTHM. Actinides - 1% of Table 3.3.10 per MTHM. Activation Products - 0.1% of Table 3.3.7 per MTHM.
7.7 – Tornado strikes mined salt storage area. Estimated frequency 8×10^{-5} km^2-yr.	1. Tornado strikes mined salt storage area. 2. Salt dispersed to atmosphere.	1. Site selection criteria limit maximum credible tornado. 2. Salt is covered as it accumulates.	Tornado strikes when pile contains maximum amount of salt. Pile is 1 km wide at bottom, 910 m wide at top, 30 m tall and 940 m long. 3.6×10^{7} MT is total salt stored in pile. Only 7×10^{4} m^2 is uncovered and available for dispersion. 2.2×10^{4} MT (1%) is removed by tornado.
7.8 – LLW drum rupture due to mechanical damage and fire.	(TRU LLW not produced in this fuel cycle. This accident is described in Section 7.5.)		
7.9 – LLW drum rupture due to internal explosion.	(TRU LLW not produced in this fuel cycle. This accident is described in Section 7.5.)		

TABLE 7.4.13. Postulated Non-Design Basis Accidents - Once-Through Fuel Cycle Repositories

Accident No. and Description	Sequence of Events	Safety System	Release
7.10 — Nuclear warfare.	1. 50-megaton nuclear weapon bursts on surface above repository. 2. Crater formed to 340 m, with fracture zone to 500 m.	1. Repository depth of 540 m - Salt 590 m - Granite 420 m - Shale 560 m - Basalt	Although the fracture zone reaches the repository in shale, releases are expected to be less than those from Accident 7.11 (repository breach by meteorite).
7.11 — Repository breach by meteorite. Expected frequency $\sim 2 \times 10^{-13}$ per year.	1. Meteor with sufficient mass and velocity to form 2 km dia crater impacts repository area after closure. 2. Crater extends to waste horizon, dispersing waste to atmosphere. 3. Crater partially refilled with rubble covering repository.	1. Repository depth of 600 m.	One percent (1%) of the inventory is released on impact with one half going to local fallout and one half going to strato-pheric dispersion.
7.12 — Repository breach by drilling No expected fre-quency can be assigned to this occurrence.	1. Societal changes lead to loss of repository records and location markers. 2. Drilling occurs.	1. Repository depth 2. Repository location monuments and records. 3. Site criteria--no desirable resources.	Drilling may occur anywhere in repository. Probability of contacting a cotaminated zone and/or canister (within 30 cms radius of canister), 0.005. One-fourth of a canister is brought to the surface in the drilling mud. The activity is uniformly distributed over 1.2 acres in the top 2 inches of soil.
7.13 — Repository breach by solution mining (salt only).	1. Societal changes lead to loss of records and location markers. 2. Exploratory drilling (see Accident 7.12) leads to the discovery of salt. 3. Salt is mined using solution extraction techniques. 4. Contamination is discovered after 1 year and mining is discontinued.	1. Monuments mark repository location. 2. Repository depth of 540 m. 3. Site criteria exclude areas with desirable resources. 4. Other plentiful and accessible salt deposits are available.	As salt is dissolved and carried away, water comes into contact with exposed spent fuel. At a leach rate of 1×10^{-5} g/cm^2 day 7 MTHM are leached into the brine during the first year of solution mining.
7.14 — Volcanism	1. Volcanic activity at repository carries wastes to surface.	1. Site criteria--no history or potential for volcanic activity.	Release equal to or less than Accident 7.15.

TABLE 7.4.13. (contd)

Accident No. and Description	Sequence of Events	Safety System	Release
7.15 – Faulting and groundwater transport. Estimated frequency 2×10^{-13} per year.	1. Fault intersects repository 2. Access is created between high-pressure aquifer, waste and surface. 3. Aquifer carries wastes to surface.	1. Site criteria--low seismic risk zone. 2. Site criteria--minimal groundwater. 3. Repository depth 4. Low leachable waste forms.	Estimated frequency of a fault intersecting the repository is 4×10^{-11}/yr. Assumed frequency that high pressure aquifer exists with canister and surface access is 0.005. The overall expected frequency is 2×10^{-13}/yr. A 12 m wide line fault is assumed to intersect the repository in an orientation to maximize waste contact. The following spent fuel MTHM are available for leaching. PWR BWR Salt 220 120 Granite 450 420 Shale 240 150 Basalt 530 280
7.16 – Erosion	1. Repository overburden subject to high erosion.	1. Site criteria--low erosion rates. 2. Repository depth.	Release equal to or less than Accident 7.11.
7.17 – Criticality	1. Fault or groundwater action brings two spent fuel canisters together.	1. Site criteria--low seismic risk zone 2. Site criteria--minimal groundwater. 3. Repository depth.	None

The accident descriptions are similar for all geologic media except for 7.7 (tornado strikes salt pile) and 7.13 (Repository breach by solution mining), which are postulated only for salt. However, the amount of spent fuel handled and emplaced does vary with the selected media. These variations are considered in the release column for the source terms.

Minor and moderate accidents considered in this section are related to the operational phase of the repository. They postulate various mechanical and thermal environments to which the canistered spent fuel may be subjected.

The release fractions for these accidents were developed using energy absorption models and conservative assumptions on waste dispersion.

Secondary wastes generated onsite are incinerated and immobilized before emplacement. Accidents for these processes are presented in Sections 4.3, 4.4 and 4.7 of this report. Releases from waste treatment accidents at this facility would be much less than those at the FRP or MOX-FFP.

Accidents for the isolation phase of the repository operation are unsophisticated in their description of geologic processess, but are believed to provide conservative estimates of radionuclide releases.

For purposes of environmental consequence analysis, the material release associated with accidents numbered 7.2, 7.6, 7.7, 7.11, 7.12, 7.13, and 7.15 in Tables 7.4.11, 7.4.12, and 7.4.13 have been selected as umbrella source terms (The concept of an umbrella source term is explained in Section 3.7 - Basis for Accident Analysis). This means that the releases from these accidents are the largest in their respective source term categories. The environmental consequences of these accidents are described in DOE/ET-0029.[5]

7.4.10 Costs For Once-Through Fuel Cycle Repositories

The methods used for developing cost estimates as well as the resulting cost estimates for repositories in four geologic media for the once-through fuel cycle are described in this section. Significant changes in methodology from that discussed in Section 3.8 are outlined here. All estimates are made in constant dollars in terms of the buying power of the dollar in mid-1976.

Estimates of construction, operating and levelized unit costs are presented here. As explained in Section 3.8, the levelized unit cost translates the construction and operating costs into an equivalent, one-time, unit charge.

7.4.10.1 Construction Costs for Once-Through Fuel Cycle Repositories

1978 construction cost estimates prepared by an architect engineering firm, Parsons, Brinkerhoff, Quade and Douglas (PBQ&D), for the repository concepts described in References 1 and 2 were used as the basis for the repository construction costs reported here. These estimates were first reviewed for reasonableness of costs and completeness by a second AE firm, Bechtel, Inc., and then revised to accommodate the CWMS waste forms, transportation assumptions, and a more conservative thermal criteria (see Section 7.3). For consistency with the cost estimates in other sections of this report the estimates were converted to a mid-1976 cost basis using 1/1.17 factor. Results are presented in Tables 7.A.1 through 7.A.8 in Appendix 7.A for all four geologic media and for both accelerated and continuous mining.

The estimates presented have an accuracy range of ±20%. This range reflects the uncertainties that are likely to be encountered during the design and construction phase of this project, but difficult or impossible to identify at this time. These uncertainties include siting and engineering scope requirements necessary to provide a fully functional facility. Also included are the possible variances of the assumed rock densities used in the development of mining costs. Not included are scope changes dictated by changes in licensing requirements or other regulatory criteria which do not now exist. Also not included is escalation because all costs are expressed in mid-1976 dollars. Within the stated accuracy range, there is an approximately equal likelihood of the indicated cost overrun or underrun.

Owners costs are included in the construction cost estimates as explained in section 3.8. However. the owners cost estimates for repositories do not include interest during construction. This cost is included as interest charges on construction expenditures prior to facility startup as part of the levelized unit cost calculation.

Table 7.4.14 summarizes the facility construction and mining costs for 2000-acre repositories in all four media and for both accelerated and continuous mining. The difference in costs between media is mostly due to larger quantities mined in granite and basalt repositories and to the higher cost per ton for mining these latter media. It should be noted, however, that because of differences in thermal loading criteria, both granite and basalt repositories can store considerably more waste (see Table 7.4.3), thus the unit cost differences are less substantial (see the subsequent section on levelized unit costs).

TABLE 7.4.14. Total Repository Construction and Mining Costs by Geologic Media for the Once-Through Fuel Cycle, millions of 1976 dollars

Mining Method and Media	Construction	Mining	Backfilling	Total
Accelerated Mining				
Salt	430	380	70	880
Granite	530	1530	180	2240
Shale	410	640	80	1130
Basalt	560	1850	220	2630
Continuous Mining				
Salt	420	380	50	850
Granite	480	1530	150	2160
Shale	390	640	70	1100
Basalt	510	1850	170	2530

The total costs in Table 7.4.14 of accelerated mining are not significantly higher than those of continuous mining in spite of the additional costs of transporting the mined material to the surface and temporarily storing it. The advantage of large scale mining operations offset the additional costs of transporting more mined material to the surface. However, since accelerated mining requires earlier expenditure of funds, the interest cost causes the levelized unit costs of the accelerated mining alternative to be significantly higher.

The construction and mining cash flow schedules were derived by applying the percentage schedules in Tables 7.4.15 through 7.4.18 to the appropriate construction expenditure. Tables 7.4.15 and 16, respectively, are schedules for constructing surface facilities, and underground facilities including shafts and hoists and underground support facilities. Table 7.4.17 is the expenditure schedule for accelerated mining in all media. Table 7.4.18 is the expenditure schedule for backfilling. The accelerated mining schedule was chosen as a reference case. This reference expenditure pattern assumes that the repository is ready to start waste emplacement operations at the beginning of year 6 (beginning of year 1 is the beginning of construction) and that the entire repository is mined out at the conclusion of year 10. Expenditure schedules for alternative continuous mining of the repository are shown in Appendix 7A in Tables 7.A.9 to 7.A.12.

TABLE 7.4.15. Expenditure Schedule for Constructing Repository Surface Facilities (All Media and Fuel Cycles)

Time (years)	Materials		Labor	
	Percent Expended	Cumulative	Percent Expended	Cumulative
1	17.6	17.6	7.0	7.0
2	13.9	31.5	21.1	28.1
3	27.1	58.6	26.8	54.9
4	29.9	88.5	26.8	81.7
5	11.5	100.0	18.3	100.0

TABLE 7.4.16. Expenditure Schedule for Constructing Repository Underground Facilities (All Media and Fuel Cycles)[a]

Time (years)	Materials		Labor	
	Percent Expended	Cumulative	Percent Expended	Cumulative
1	25.4	25.4	5.0	5.0
2	25.6	51.0	16.5	21.5
3	19.6	70.6	28.9	50.4
4	18.2	88.8	30.0	80.4
5	11.2	100.0	19.6	100.0

a. Includes shafts, hoists and underground support facilities.

TABLE 7.4.17. Expenditure Schedule for Repository Mining (All Media and Fuel Cycles - Accelerated Mining Only)[a]

Time (years)	Materials		Labor	
	Percent Expended	Cumulative	Percent Expended	Cumulative
1	--	--	--	--
2	0.4	0.4	--	--
3	8.3	8.7	7.5	7.5
4	13.1	21.8	13.6	21.1
5	14.4	36.2	13.6	34.7
6	13.8	50.0	13.6	48.3
7	13.6	63.6	13.6	61.9
8	13.4	77.0	13.6	75.5
9	13.0	90.0	13.6	89.1
10	10.0	100.0	10.9	100.0

a. Includes mining of main corridors and emplacement rooms.

TABLE 7.4.18. Repository Expenditure Schedule for Backfilling and Shaft Sealing

Media	Fuel Cycle	Years	Materials and Labor Percent per Year
Salt	Once-Through	11	9.1
Shale	Once-Through	13	7.7
Granite	Once-Through	20	5.0
Basalt	Once-Through	20	5.0

Construction costs were allocated to waste type (PWR and BWR) based on data tabulated in Appendix 7C in Tables 7.C.1 through 7.C.8. Canistered waste costs, nonallocable costs and shaft costs were further allocated to each waste type (PWR and BWR) by developing ratios based on the number of canisters of each waste type received to total canisters received. These ratios were applied to the construction cash flows previously developed to derive the waste-specific construction and mining cash flows.

7.4.10.2 Operating Costs for Once-Through Fuel Cycle Repositories

Operating costs for the repository were developed as a function of waste type and year to take into account changing waste receipt rates with time. Operating costs are defined to include direct labor, operating materials, utilities, hole drilling and/or trenching, sleeve and canister emplacement, overhead and contingency costs.

Schedules of personnel requirements for waste storage operations and waste receipt rates from Y/OWI/TM-36/16[6] were correlated using polynomial curve fitting techniques to derive personnel requirements as a function of total waste receipt rate for the reference waste types. The personnel were allocated by job category using data for a reference year from the same reference.[6] Table 7.4.19 shows the job categories with the percentage allocations of variable operating personnel in each category for the four geologic media. Annual personnel requirements were based on the general operations category listed in Table 7.4.7 less a constant administrative work force of 110 people. Personnel in each job category were allocated by waste (spent fuel) type using job-specific allocation factors based on relative canister receipt rates, relative mine volumes for waste placement and other factors. Wage rates were developed using weighted averages of wage rates for skills in each job category based on the standard skilled labor salary schedules given in Section 3.8 plus 30% for fringe benefits and other direct overhead. Total labor costs by waste type were then derived by summing the labor costs by waste type over all job categories. Labor costs associated with hole drilling, backfilling and sleeve emplacement are not included above, but are included separately in the unit cost calculation for these services as described below.

Operating materials are defined to include overpacks, retrievability sleeves and plugs and waste emplacement racks for trenching operations. Operating materials costs are based on the requirements shown in Table 7.4.8 (plugs and retrievability sleeve requirements assume a 5-year readily retrievable period) and the unit costs for each container type given in Table 7.4.20. Emplacement of BWR canisters in trenches requires a sleeve for each canister and a rack holding eight sleeves throughout the entire repository operation.

TABLE 7.4.19. Percentage of Variable Operating Personnel[a] by Job Category and Repository Type - Once-Through Cycle

Job Category	Percent of Total Direct Labor Force Salt	Granite	Shale	Basalt
Underground Storage	9.7	12.2	10.9	12.5
Above Ground Storage	22.3	27.8	25.0	28.8
Control Room	3.2	3.8	3.6	3.9
Railroad Operations	6.0	5.8	6.2	5.9
Shipment Inspections	2.0	2.0	1.9	2.0
Liquid Rad Waste	1.8	2.4	2.0	2.3
Ventilation Maintenance	1.3	1.1	1.2	1.0
Utilities Support	2.3	1.9	2.2	1.8
Motor Pool	0.9	0.7	0.8	0.7
Maintenance	6.4	5.2	5.9	4.9
Security	15.1	12.1	13.9	11.4
Supervision	18.5	16.7	16.8	16.6
Other	10.5	8.3	9.6	8.2
	100.0	100.0	100.0	100.0

a. Variable operating personnel are determined as the work force listed in column 1, Table 7.4.7 less a constant administrative workforce of 110 persons.

TABLE 7.4.20. Unit Costs for Overpacks, Racks, Sleeves and Plugs - Once-Through Cycle Repositories

Description	Unit Cost, 1976$
PWR Overpacks	550
BWR Overpacks	450
Retrievability Sleeve and Plug for PWR Canister	930
Retrievability Sleeve and Plug for BWR Canister	730
Trench rack for 8 BWR canisters	4000

Utilities costs were estimated from requirements given in Table 7.4.9 and the unit utilities costs given in Section 3.8. These costs were allocated by waste type using relative mined volumes attributable to each waste.

Hole drilling costs were developed using annual canister emplacement requirements (Table 7.4.1) and the estimated unit hole drilling costs shown in Table 7.4.20. For granite and basalt repositories the BWR spent fuel elements are placed in sleeves in backfilled trenches. The unit cost of trenching is shown in Table 7.4.21. Unit costs of emplacing sleeves in holes or trenches is shown in Table 7.4.22. The unit man-hour requirements associated with each operation are also given in Table 7.4.21 and 22.

TABLE 7.4.21. Unit Hole Drilling and Trenching Costs - Once-Through Cycle Repositories

Media	Operation	Cost Per Canister, 1976 $		Man-hours per Canister	
		PWR	BWR	PWR	BWR
Salt	Hole Drilling	380	340	7	6
Granite	Hole Drilling	3750	--	49	--
	Trenching	--	430	--	6
Shale	Hole Drilling	800	730	14	13
Basalt	Hole Drilling	3940	--	53	--
	Trenching	--	440	--	6

TABLE 7.4.22. Unit Sleeve Emplacement Costs - Once-Through Cycle Repositories

Operation	Cost per Canister, 1976 $	Man-hours per Canister
Sleeve Emplacement	130	4

Administrative overhead requirements were based on requirements given in Y/OWI/TM-36/16[6] and amount to a work force of approximately 110 persons. Wage rates for technical and administrative personnel were based on wage rates for these personnel in nuclear operations given in Section 3.8. Overhead labor is allocated to waste type in the same ratio as direct labor.

Total operating staff manpower requirements, representing the sum of direct, administrative, hole drilling, emplacement and backfill labor, were given in Table 7.4.7.

An additional 25% contingency allowance for miscellaneous and unidentified costs was added to all of the operating cost estimates.

Table 7.4.23 tabulates total operating costs (in 1976 dollars) over the entire repository lifetime by expenditure type and repository media for the once-through fuel cycle.

TABLE 7.4.23. Total Operating Costs for Spent Fuel Repositories in Millions of 1976 Dollars

Operating Expenditure	Salt	Granite	Shale	Basalt
Direct Labor	190	375	240	350
Materials	40	315	30	315
Utilities	60	110	70	110
Hole Drilling/Trenching	60	715	160	750
Sleeve Placement	5	35	5	35
Overhead	50	70	50	70
Contingencies	100	400	135	410
TOTAL	505 ±25%	2,020 ±25%	690 ±25%	2,040 ±25%

The largest operating cost item in salt and shale repositories is the direct labor cost with the other components costing smaller but significant amounts. In granite and basalt repositories the hole drilling and trenching costs overshadow all other operating costs and comprise about 55% of the total. The large differences in hole drilling and direct labor costs between the salt-shale and granite-basalt repositories are due to the greater difficulty in drilling the latter media and the greater amount of waste stored in basalt and granite (i.e., the repositories operate longer). The higher materials costs in basalt and granite stem from the use of sleeves and racks in the BWR trenches.

7.4.10.3 Backfilling Costs for Once-Through Repositories

Since backfilling costs are treated as a construction contract, they are listed as line items in the construction costs tables in Appendix 7A. They also appear in Table 7.4.14. The expenditure schedule for backfilling was previously listed in Table 7.4.18 and assumes backfilling operations beginning in the eleventh year after construction startup (i.e., beginning in the sixth year after startup of waste emplacement operations).

7.4.10.4 Decommissioning Costs for Once-Through Fuel Cycle Repositories

Decommissioning costs are defined to include decommissioning of the surface facilities and shaft sealing costs. The decommissioning cost of the surface facilities is estimated at 10% of the construction cost of these facilities. Shaft sealing costs are shown in Tables 7.A.1-7.A.8 in Appendix 7A. Decommissioning and shaft sealing costs for the different repositories are tabulated in Table 7.4.24.

TABLE 7.4.24. Decommissioning and Shaft Sealing Costs for Once-Through Fuel Cycle Repositories

	Mid-1976 $1,000,000
Salt	21.6
Granite	21.1
Shale	21.4
Basalt	21.0

7.4.10.5 Levelized Unit Costs for Once-Through Fuel Cycle Repositories

Levelized unit costs for PWR and BWR spent fuel stored in the once-through cycle repository are shown in Table 7.4.25 for the reference accelerated mining case. Unit costs by waste type are calculated by identifying the construction and operating cost components attributable to each waste type, developing the cash flows, and using the levelized unit cost procedure given in Section 3.8. Waste quantities received are based on a 400-GWe nuclear power system in year 2000 and are shown in Tables 7.4.1 and 7.4.3.

TABLE 7.4.25. Levelized Unit Cost Estimate for Spent Fuel Repositories, Accelerated Mining, \$/kg HM[a]

Geologic Media	Waste Type	Construction and Mining Cost[d]	Operating Cost	Total Unit Cost[b,c]	Total Unit Cost for 0 to 10% Cost of Capital
Salt	PWR	32.40	9.90	42.20	26 - 52
	BWR	35.00	12.50	47.50	30 - 58
	average	33.40	10.90	44.30	28 - 54
Granite	PWR	46.80	17.20	64.00	34 - 84
	BWR	50.10	20.00	70.10	34 - 89
	average	48.10	18.40	66.50	34 - 86
Shale	PWR	33.60	9.80	43.40	26 - 54
	BWR	40.70	16.30	57.00	35 - 71
	average	36.30	12.30	48.60	29 - 60
Basalt	PWR	54.80	17.30	72.10	37 - 96
	BWR	58.40	19.40	77.80	37 - 99
	average	56.20	18.10	74.30	37 - 97

a. The costs may be converted to \$ per container by multiplying PWR costs by 461 kg/container and BWR costs by 189 kg/container.
b. Assuming a cost of money of 7% to the Federal government.
c. Overall uncertainties for all total levelized unit costs are estimated to be $\pm 50\%$.
d. Includes backfilling and decommissioning costs.

Construction and mining costs account for the bulk of the unit costs comprising about 75 percent of the total. Geologic disposal of spent fuel in salt appears to be slightly cheaper than in shale although because of the uncertainties involved the advantage is marginal. Salt and shale disposal appear to have a distinct cost advantage over granite and basalt mainly due to the expense of mining and drilling in these more difficult media. Since costs are on a unit basis, the greater capacities of the granite and basalt repositories are taken into account in the above cost comparisons. The last column in Table 7.4.25 illustrates the effect on levelized unit costs of using costs of money to the Federal government of 0 and 10 percent (the reference case is 7%). Since construction and mining, which account for most of the cost, are completed in the first five years of operation in the accelerated mining case, the cost of money has a substantial effect on the unit cost. A cost of money of 0 percent would reduce unit costs from 35 to 50 percent and a cost of money of 10 percent would increase unit costs 20 to 35 percent.

The levelized unit costs corresponding to the alternative continuous mining case are shown in Table 7.4.26. Since mining costs are spread over the entire operating period, revenues match expenditures more closely, interest charges are reduced and the levelized unit cost is reduced for all media. However, since the mining costs in granite and basalt are a much greater

TABLE 7.4.26. Levelized Cost Estimates for Spent Fuel Repositories - Continuous Mining, $/kg HM

Geologic Media	Waste Type	Levelized Construction Cost	Levelized Operating Cost	Total Levelized Unit Cost
Salt	PWR	27.90	9.80	37.70
	BWR	31.20	12.60	43.80
	average	29.20	10.90	40.10
Granite	PWR	33.40	17.20	50.60
	BWR	36.20	20.00	56.20
	average	34.50	18.30	52.80
Shale	PWR	27.40	9.80	37.20
	BWR	34.30	16.20	50.50
	average	30.00	12.20	42.20
Basalt	PWR	38.80	17.30	56.10
	BWR	41.30	19.40	60.70
	average	39.80	18.10	57.90

fraction of total construction costs than they are in salt and shale, these costs are reduced relatively more and the overall dollar cost differential between media is substantially less even though the ranking does not change.

Taking the unit costs for accelerated mining as the reference case (Table 7.4.25) and considering the uncertainty in the mining procedure and in the cost of money, the overall uncertainty including the uncertainties in capital and operating costs was estimated to be on the order of $\pm$50%.

The costs of a 25-year ready retrievability period for a repository in salt were briefly examined (see Appendix 7D for further discussion) and were estimated to be 74 $/kg HM for PWR fuel and 84 $/kg HM for BWR fuel. (Incremental costs of $31 and $34/kg HM respectively.) The higher costs are primarily due to decreased repository capacity resulting from the more restrictive thermal criteria.

If retrieval of spent fuel elements from the repository is necessary during the 5-year readily retrievable period the costs are estimated to be no more than:

	Mid-1976 $/kg HM			
	Salt	Granite	Shale	Basalt
Retrieval	12	15	13	15
Interim Storage	19	19	19	19
Shipment to New Repository (1500 mi)	27	27	27	27
TOTAL	58	61	59	61

These cost estimates are based on: a) retrieval costs equal to emplacement costs; b) interim storage costs based on dry caisson storage costs (see Section 5.7.7), and c) shipment costs based on costs developed in Section 6.2. The cost of retrieval after backfilling was not developed here.

7.4.11 Construction Requirements For Once-Through Fuel Cycle Repositories

Many of the activities relating to site preparation and facility construction may have some impact on the environment, the local economy, and the natural resources surrounding the area. The information which follows provides a basis for evaluating the impact of construction activities. These activities include construction of surface facilities, shafts and all mining.

7.4.11.1 Construction Activities

The engineering and construction schedule for this project is dependent on site evaluation and acquisition, federal approvals, funding, and other activities. The schedule is further complicated by technical considerations including the coordination of underground and surface construction activities. For this report it is assumed that an environmental impact statement and *Preliminary Safety Analysis Report* are prepared and approved in a procedure similar to that required for a commercial fuel-cycle facility. It is also assumed that a suitable site has been selected and subjected to geologic and safety evaluations before construction operations start and that no unanticipated underground conditions are encountered in the course of shaft sinking and mining. Figure 7.4.14 summarizes the overall project schedule for engineering, procurement, and construction phases of the project.

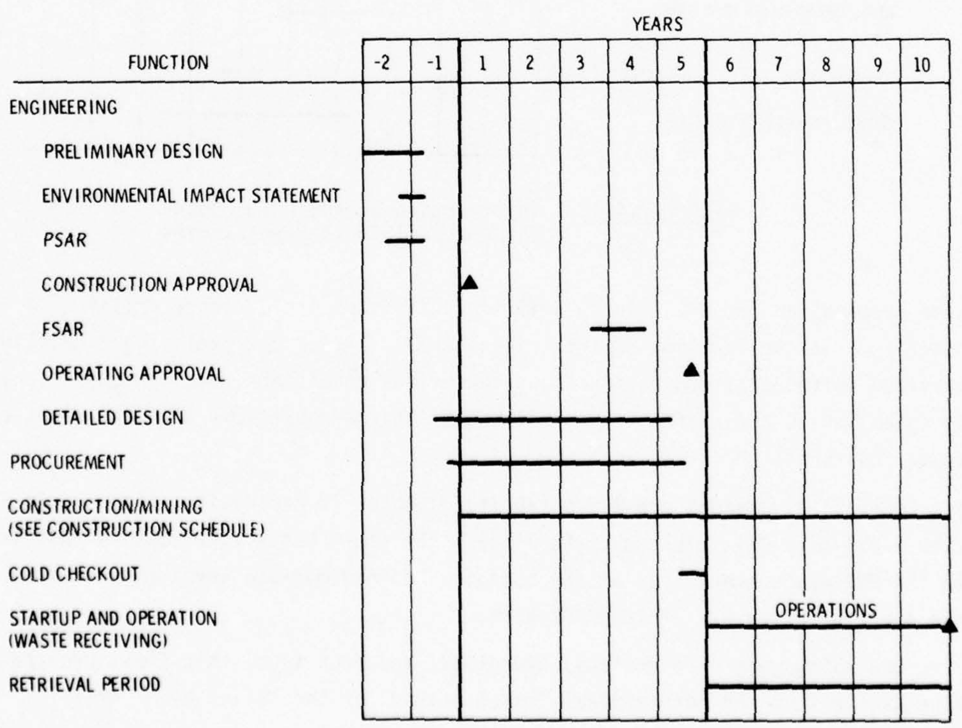


FIGURE 7.4.14. Engineering and Construction Schedule for Once-Through Fuel Cycle Repositories

The shaft and underground construction schedule is shown in Figure 7.4.15. The construction phase of the underground radioactive waste disposal facilities is performed in the following sequence:

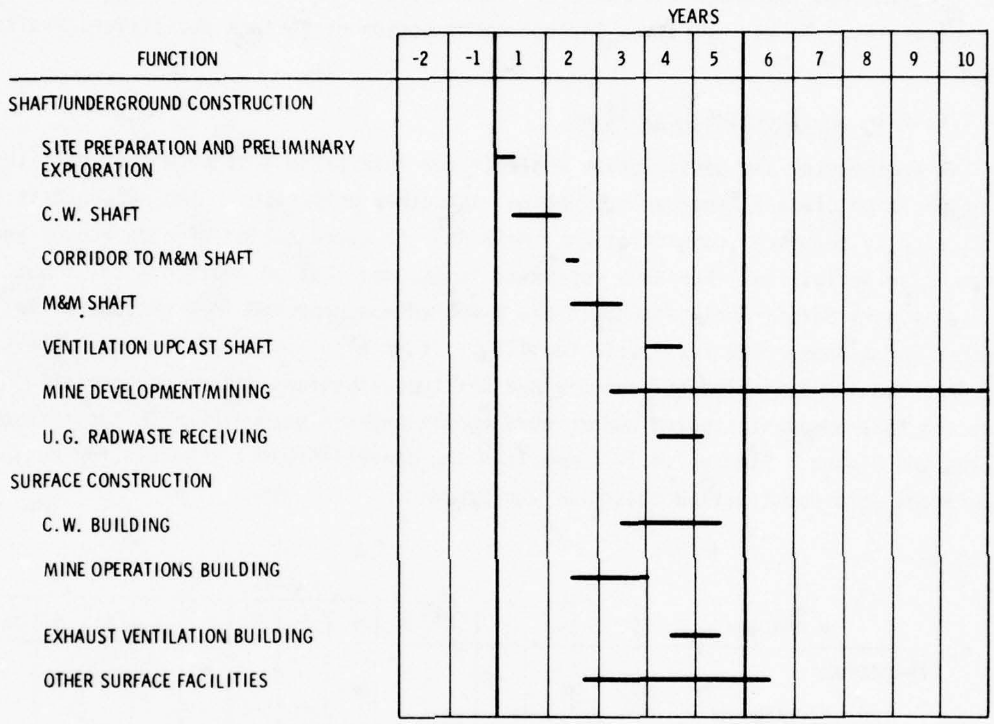


FIGURE 7.4.15. Construction Schedule for Once-Through Fuel Cycle Repositories

- Site preparation and preliminary drilling activities include core drilling to verify conditions at the location of the initial shaft. After the preliminary drilling has provided sufficient information about subsurface conditions, and the shaft location is established, the shaft-sinking contractor starts operations with the canistered waste (CW) shaft.
- The CW shaft dimensions are 4.9 m (16 ft) diameter in excavation, finished to 4.3 m (14 ft) with a 0.3-m (1-ft) thick concrete liner. The shaft has a mine station that is connected to the PWR emplacement area at the bottom. An intermediate level mine station connects the shaft with the BWR emplacement area.

Depending upon site geology, hydrology, and rock type, this first penetration to the mine uses either the conventional "bench method" or the "blind hole" drilling method for shaft sinking.

In the bench method, the bottom of the shaft is advanced using hand-held drills to drill holes over one-half the shaft area. The holes are loaded with explosives, and the rock blasted. Alternative halves of the shaft are drilled and blasted to advance the entire shaft bottom. When water bearing strata are approached, holes are drilled from the shaft bottom into the water bearing zone. Packers are installed in these holes and grout is injected to seal the rock formation in the region of the shaft. The concrete shaft lining is installed after each 12- to 18-m (40- to 60-ft) advance.

The blind hole drilling method uses highly specialized equipment and techniques. Shaft drilling differs from ordinary rotary drilling in the way force is applied to the bit and in the technique used to remove the cuttings.

In the blind hole method, drilling pressure is maintained by adding heavy "parasite" weights to the drill string immediately above the bit. Drill cuttings are removed by a method called "reverse circulation air lift". Using this method, the drilling fluid is added to the annular space and the cuttings are removed up the inside of the drill string which is the reverse of ordinary rotary drilling. This technique is illustrated in Figure 7.4.16.

After drilling is completed, the drill rods, weights, and bit are removed from the hole, which is left full of drilling mud. The shaft liner, consisting of an outer shell of steel and an inner shell of concrete, is then lowered into the shaft in prefabricated 3-m (10 ft) lengths.

Temporary services and a skip hoist are then installed to support underground operations while succeeding shafts are being sunk.

- From the CW shaft BWR level mine station, a corridor is driven toward the men and materials (M&M) shaft. At the end of the corridor (the bottom of the future shaft) an assembly room is excavated where the raise boring bit is assembled. From the surface, a directional hole is drilled to assure a vertical shaft. The drill rods required to upream the shaft are lowered through this hole. The M&M shaft dimensions are 8.8 to 9.4 m (27 to 31 ft) diameter, with 0.3 m (1 ft) minimum of concrete lining. The M&M shaft cannot be upreamed in one stage because of its dimensions. Therefore, a 1.8-m (6-ft) excavation diameter shaft is first raise-bored within the M&M shaft. Excavation enlargement and the concrete liner placement are performed by slashing the walls downwards. The muck is removed from the bottom of the shaft through a chute and lifted to the surface by the temporary CW shaft hoist.

  A light-weight headframe with a small hoist is used to lower men and materials during the sinking of the shaft. Following completion of the M&M shaft, the permanent headframe and skip hoist are installed and use of the temporary services in the CW shaft is discontinued.

- At the granite and basalt repositories a mine production (MP) shaft is constructed with similar techniques at the same time as the M&M shaft. This shaft is 8.8 to 9.1 m (29 to 30 ft) in dia and contains skip hoists for removal of mined rock.

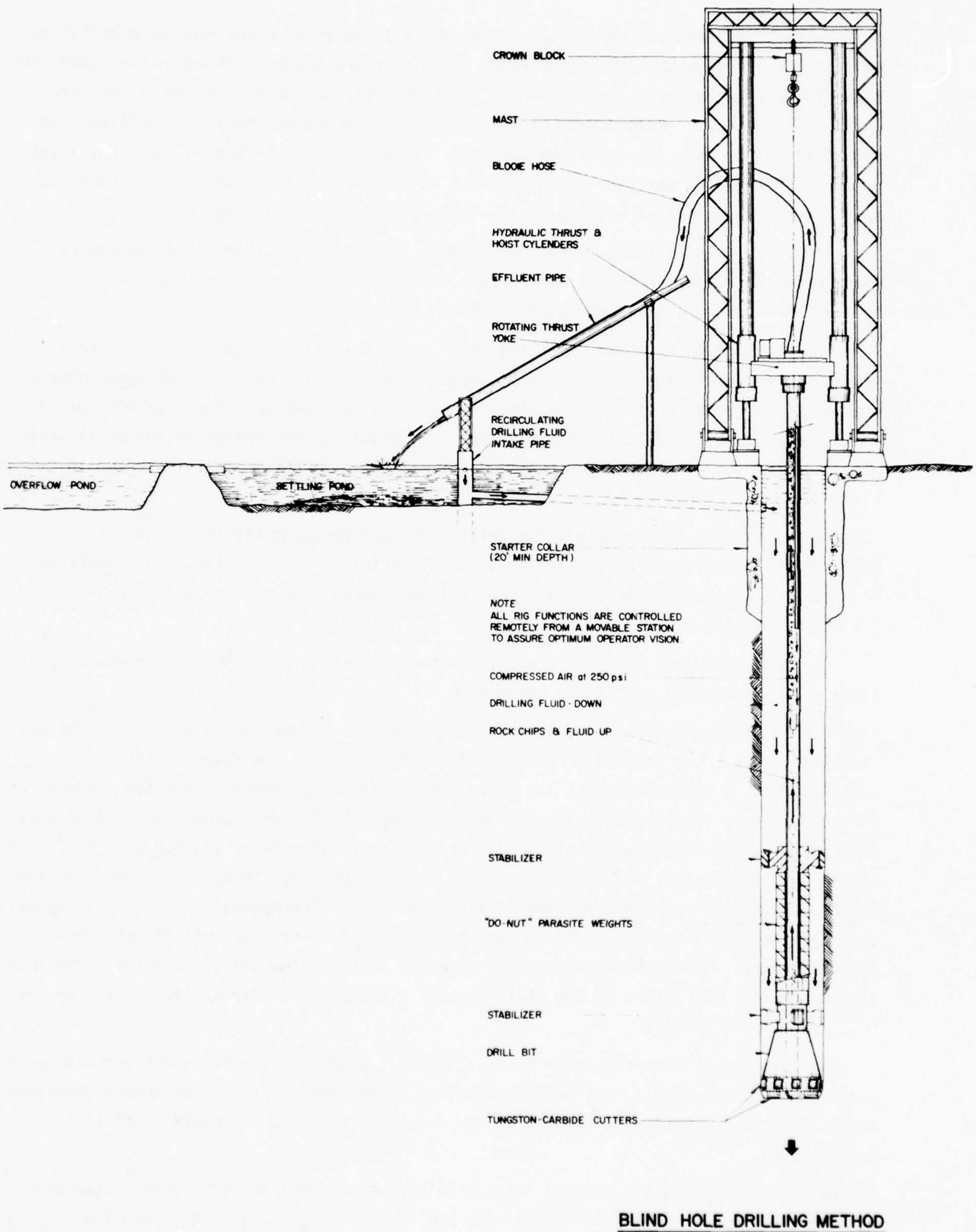


FIGURE 7.4.16. Blind Hole Drilling Method

- The ventilation exhaust shaft is constructed in a sequence similar to that for the M&M shaft. The shaft is bored to 8.5-9.1 m (28-30 ft) dia and lined with 0.3 m (1 ft) of concrete. No hoist is required.
- From each shaft the subcontractor drives the main mine level corridors to a distance of 61 m (200 ft). This distance is sufficient to erect the mining machinery and begin development of the central underground facilities including those underground receiving facilities for radioactive waste.
- For a repository in salt, the mining operations are performed using electrically powered continuous mining units that cut an opening 3.4 m (11 ft) high x 5.5 m (18 ft) wide. On a second pass, the mining machine enlarges the opening to 3.4 m high x 11 m (36 ft) wide. Two later passes in the floor produce a corridor with the dimensions of 6.7 m (22 ft) high x 11 m wide. Thus, a total of 4 machine passes are required. The 6.7-m high rooms require floor cutting, which causes unbalanced pressures on the mining machine rotors and increases maintenance costs. For 5.5-m wide x 6.7-m high rooms, only two passes are required.

  Extendible conveyors in each room transfer the mined salt to roof-suspended main line conveyor systems that transport the salt to an underground bin. A reclaim conveyor beneath the storage bins is fed by automatically controlled vibratory feeders. The salt is held in a surge bin before being conveyed to a skip hoist for removal to the surface.
- Conventional drilling and blasting techniques are used to mine all corridors, rooms, service areas, and ventilation corridors for repositories located in shale, basalt and granite. A mining cycle consists of drilling the drift face, loading the holes with explosives, blasting, and, after allowing the fumes to clear, muck removal. Drilling is performed by jumbos equipped with electric hydraulic drills. The drilling crew drills between 60 and 90 5.1-cm (2-in.) dia holes per face, depending on the size of each face. The hole length is 3.3 m (11 ft), which allows breaking of 3.0 m of ground. When the drilling of a face is completed, the drill crew moves to a second face and commences work while preparations for blasting are being made at the first face.

  At the first face drilled, a blasting crew loads the holes with a round of explosives, and the round is blasted. While the fumes clear, the blasting crew moves on to another face that already has been drilled.

  After fumes have cleared, a mucking crew begins removing the blasted rock. The mucking crew employs a rubber-tired front-end loader equipped with a 6-yard rock bucket to load the broken rock into 35-ton (22-yard) dump-trucks. When mining the rooms, the rock is sent down the branch corridor toward the main corridor system. A grizzly located at the end of the branch corridor (just off to the side of the main corridor) uses a hydraulic breaker to reduce the size of the material. All the broken rock passing through the grizzly is dropped into a control chute through a vertical opening excavated in the rock. When mining corridors and ventilation drifts, the broken rock is dropped into the nearest chute.

From the control chute, the broken rock is fed into rail cars and transported to a centrally located rock crusher. The crushed rock, maximum 0.2-m (8-in.) material, is fed onto a conveyor located below the crusher discharge. The conveyor carries the crushed rock up a 25% incline slope and discharges it into a storage bin located near the M&M shaft, which acts as a surge control for hoisting operations. The broken rock is fed onto two short conveyor belts located directly below the storage bin. These belts feed the skip hoists in the men and materials shaft.

Following rock removal, a fourth crew may be required for bolting and scaling. The bolting crew uses a mobile bolting platform to install rock bolts in the roof and walls in a 1.2-m (4-ft) x 1.2-m pattern. Crews following the rock bolting crews install pipe lines and ventilation tubing, and maintain the roadways. Each crew moves on to another face upon finishing its task. This allows advances on a number of faces simultaneously. This process of multiple heading advance is only applied during the full mining stages.

The construction phase of the surface facilities includes:

- Permanent M&M shaft headframes and hoists and mined materials handling facilities. The mine operations building is also included. Construction of these facilities proceeds concurrently with the underground work.
- Construction of the canistered waste building, low-level waste building and mine exhaust filter building begin upon completion of the corresponding shaft sinking operations and the discontinuance of the use of the CW shaft for temporary mining operations. The construction sequence for these operations is shown on the Construction Schedule (Figure 7.4.14).

7.4.11.2 Construction Labor Requirements

The construction and mining labor requirements are estimated to be

	Man-Hours, 1000s			
	Salt	Granite	Shale	Basalt
Manual	16,244	50,414	23,362	60,803
Non-manual	3,566	11,066	5,128	13,347
Total	19,810	61,480	28,490	74,150

These man-hours are distributed over the five-year construction period and the first five years of operation as shown in Figures 7.4.17 through 7.4.20 for repositories in salt, granite, shale, and basalt. The annual construction labor force may be determined on the basis of 2000 man-hrs per man-yr.

7.4.11.3 Distribution Between Onsite and Offsite Costs

Onsite costs are those for all construction, materials and services provided at the site of the repository while offsite costs are those for all services provided, equipment fabricated and/or assembled, and material purchased offsite of the repository. The distribution of total construction and mining costs in these categories is as shown below:

	Facility Costs (mid 1978 $1000s)			
	Salt	Granite	Shale	Basalt
Onsite	268,000	686,000	344,000	802,000
Offsite	612,000	1,554,000	786,000	1,828,000
Total	880,000	2,249,000	1,130,000	2,630,000

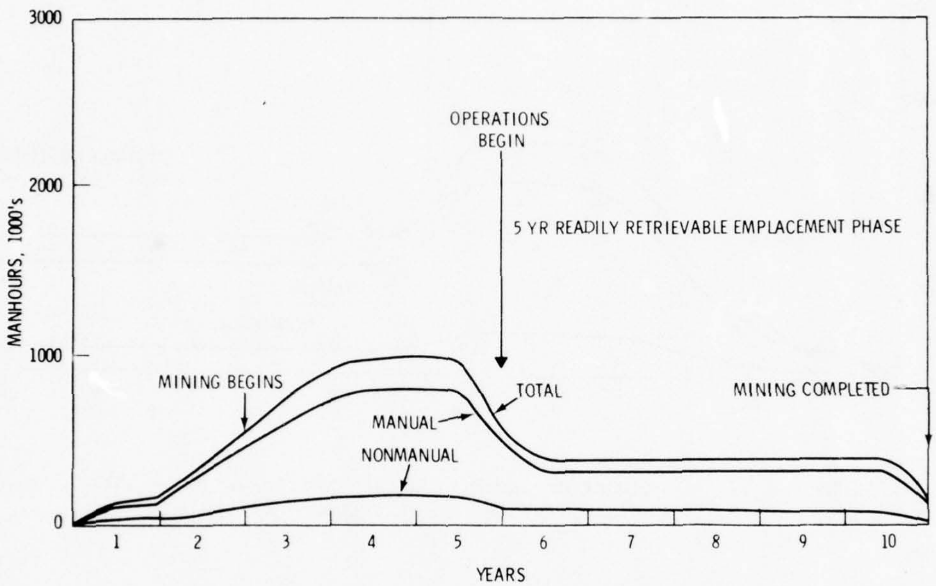


FIGURE 7.4.17. Construction and Mining Labor Distribution - Salt Repository for the Once-Through Fuel Cycle

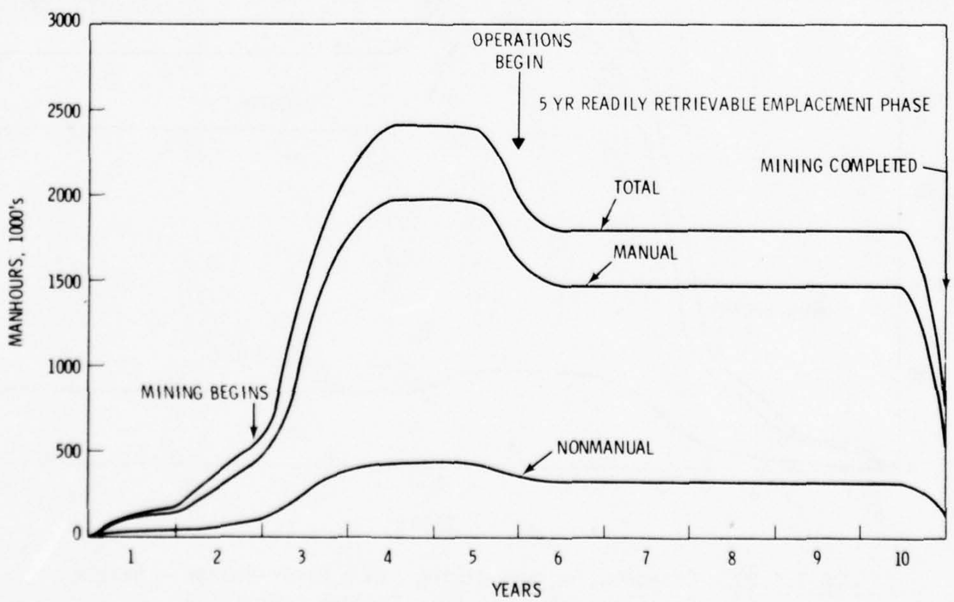


FIGURE 7.4.18. Construction and Mining Labor Distribution - Granite Repository for the Once-Through Fuel Cycle

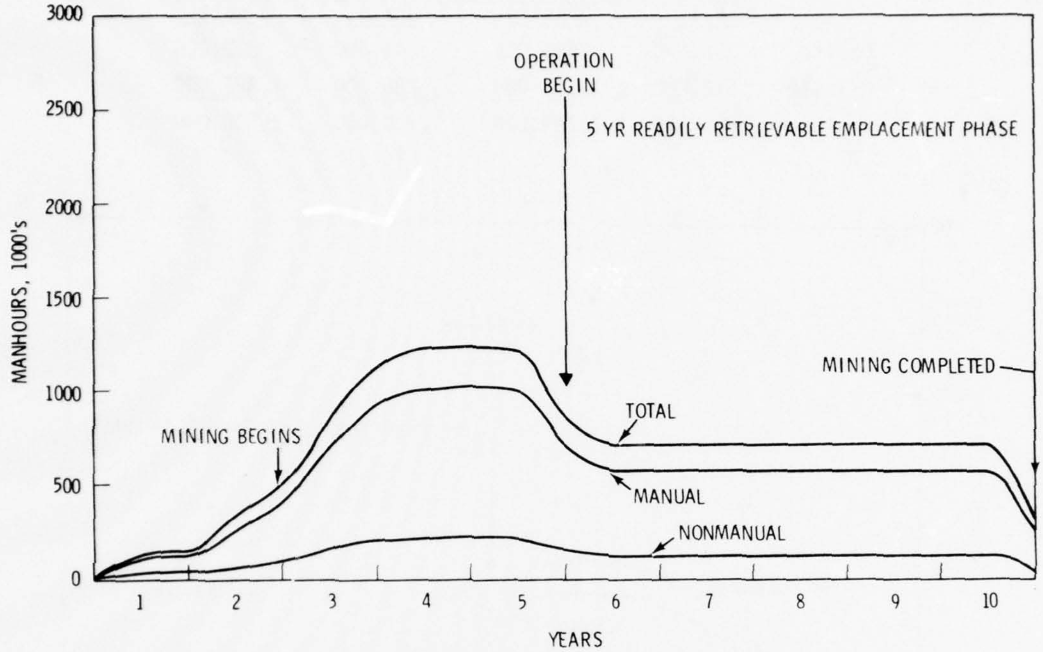


FIGURE 7.4.19. Construction and Mining Labor Distribution - Shale Repository for the Once-Through Fuel Cycle

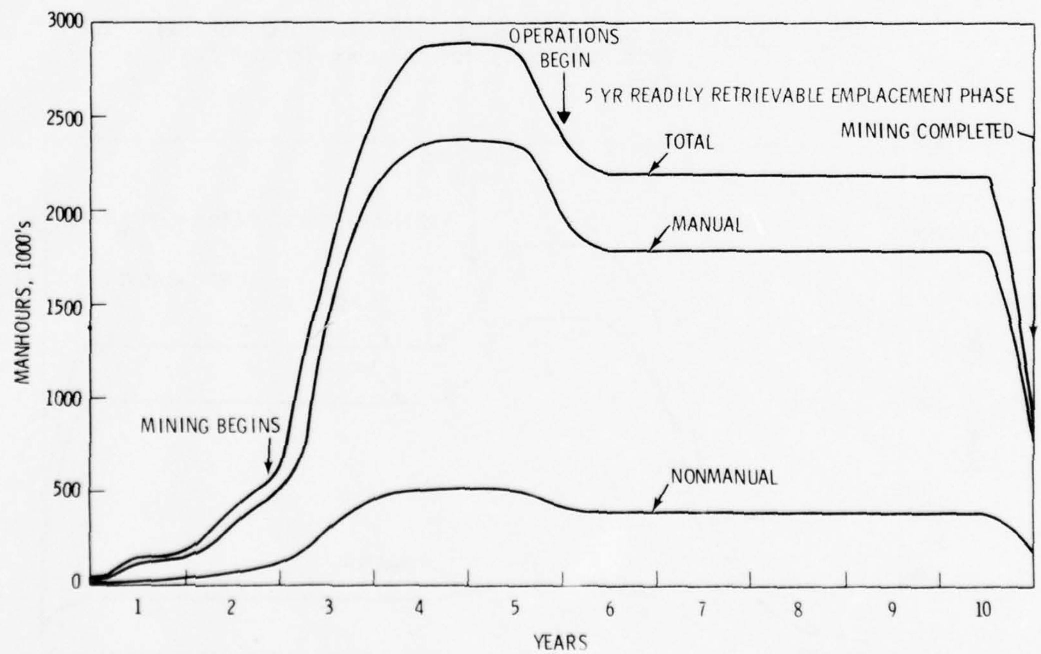


FIGURE 7.4.20. Construction and Mining Labor Distribution - Basalt Repository for the Once-Through Fuel Cycle

7.4.11.4 Resources Committed

Resources committed to construction of the repositories are listed in Table 7.4.27.

TABLE 7.4.27. Resource Commitments

	Salt	Granite	Shale	Basalt
Land Use				
Surface facilities, ha	180	280	180	280
Access roads and railroads, ha	8	8	8	8
Mineral and surface rights, ha (fenced restricted area)	810	810	810	810
Additional land on which only subsurface activities will be restricted, ha	3,200	3,200	3,200	3,200
Water Use, m^3	240,000	710,000	360,000	610,000
Materials				
Concrete, m^3	100,000	300,000	150,000	250,000
Steel, MT	16,000	48,000	24,000	40,000
Copper, MT	220	660	330	560
Zinc, MT	55	160	80	140
Aluminum, MT	41	120	64	110
Lumber, m^3	2,300	6,900	3,000	5,900
Energy Resources				
Propane, m^3	2,200	6,400	3,200	5,400
Diesel fuel, m^3	22,000	64,000	32,000	54,000
Gasoline, m^3	16,000	47,000	21,000	40,000
Electricity				
Peak demand, kW	3,400	11,000	5,100	8,800
Total consumption, kWh	14,000,000	43,000,000	21,000,000	36,000,000
Manpower, man-hours	2.0×10^7	6.1×10^7	2.8×10^7	7.4×10^7

7.4.11.5 Transportation Requirements

A railroad spur track about 3.2 km (2 miles) long must be brought into the site of the repository. This track is routed and constructed to suit the local terrain and is ultimately used for bringing the radioactive waste containers transported on rail cars.

REFERENCES FOR SECTION 7.4

1. Union Carbide Corporation, Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Y/OWI/TM-44, Office of Waste Isolation, Union Carbide Corporation, Nuclear Division, Oak Ridge, TN, 1978.

2. Union Carbide Corporation, Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations, Y/OWI/TM-36, Office of Waste Isolation, Union Carbide Corporation, Nuclear Division, Oak Ridge, TN, 1978.

3. Union Carbide Corporation, Evaluation of Salt and Mine Rock Disposal, Y/OWI/SUB-76/16507, Office of Waste Isolation, Union Carbide Corporation, Nuclear Division, Oak Ridge, TN, 1976.

4. R. L. Bradshaw and W. C. McClain, Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Waste in Underground Salt Mines, ORNL-4555, Oak Ridge National Laboratory, Oak Ridge, TN, April, 1971.

5. Environmental Aspects of Commercial Waste Management, DOE/ET-0029, Department of Energy, Washington, DC, in press.

6. Union Carbide Corporation, Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations, Y/OWI/TM-36/16, Office of Waste Isolation, Oak Ridge, TN, April, 1978. Tables 1-2 through 1-4 and Tables 5-4 through 5-8 in volumes 10, 12, 14, and Tables 1-2 through 1-13 in volume 16. See also Appendix B in volumes 10, 12, 14, and 16.

7.5 GEOLOGIC REPOSITORIES FOR THE REPROCESSING FUEL CYCLES

7.5 GEOLOGIC REPOSITORIES FOR THE REPROCESSING FUEL CYCLES

Three reprocessing fuel cycles are considered in the conceptual repository designs: uranium-only recycle with plutonium in the high-level waste (cycle IIA), uranium-only recycle with PuO_2 stored for future use or disposal (Cycle IIB) and uranium and plutonium recycle (Cycle III). A repository operating in support of any of these reprocessing fuel cycles is required to receive high-level wastes (HLW), fuel residue waste (FRW), and intermediate- and low-level transuranic (TRU) wastes. The following sections describe conceptual repositories located in salt, granite, shale, and basalt formations for the reprocessing fuel cycles. These descriptions are based on conceptual repositories that are described in References 1 and 2 and that are modified to accommodate the waste forms described in this report. These concepts do not necessarily represent an optimum design but are representative of what could be achieved with current technology. In actual applications it is reasonable to expect that there could be some improvement over these concepts that might be reflected in either more efficient operation or lower environmental impacts or both. These conceptual descriptions provide a reasonable basis for cost analysis and for development of environmental impacts.

7.5.1 Alternatives for Reprocessing Fuel Cycle Repositories

Alternatives to the conceptual repository designs presented in the sections that follow are the same as discussed in Section 7.4.1 for the spent fuel repositories.

7.5.2 Design Bases For Reprocessing Fuel Cycle Repositories

The design basis for the conceptual reprocessing fuel cycle repositories was as follows:

- The conceptual repositories are designed to receive and emplace HLW, FRW, ILW, and LLW whose characteristics are as described in Section 4. Repository layouts and capacities are based on the thermal criteria described in Section 7.3 and on reference 6.5-year old (time out of reactor) HLW, although initially the HLW is older than the reference 6.5 years because of delays prior to reprocessing. Other wastes are shipped to the repository as they are packaged.
- Canisters of HLW, FRW, and ILW are shipped by rail in shielded shipping casks and containers. ILW packaged in 55- and 80-gal drums and low-level wastes are shipped by truck. Details of waste transportation are provided in Section 6. Table 7.5.1 describes waste packages.
- The allowable thermal density for emplacement of wastes is conservatively set at two-thirds of the calculated maximum acceptable density. Additional details on thermal criteria are provided in Section 7.3.
- Overall underground area at the repositories in salt, granite, shale, and basalt is approximately 800 ha (2000 acres). Maintaining the same size repository in each rock medium results in different repository storage capacities (total equivalent MTHM emplaced) for each medium because of each medium's different thermal criteria for emplacing waste (Section 7.3).

TABLE 7.5.1. Waste Packages

Waste Type	Container	Material	Remote Handling Required
HLW	Canister 30 cm dia x 3.1 m long	Stainless steel	Yes
FRW	Canister 76 cm dia x 3.1 m long	Carbon steel	Yes
ILW	Canister 76 cm dia x 3.1 m long	Carbon steel	Yes
ILW[a]	50 and 80 gal drums	Carbon steel	Yes
LLW	1.2 m x 1.8 m x 1.8 m boxes	Carbon steel	No
LLW[b]	55 gal drums	Carbon steel	No

a. ILW drums are placed, three each, into steel drum pack canisters for further handling and emplacement.
b. LLW drums arrived on pallets of 12 drums each.

- The repositories operate in a readily retrievable mode for about five years. During this period of ready retrievability, rooms are left open (no backfill), and all emplacement holes and trenches are provided with steel liner sleeves and concrete plugs. Emplacement rooms and pillars are designed to accommodate the effects of decay heat from the wastes during this period and remain open and accessible. In salt, accelerated creep closure of the rooms is taken into account.
- All mining operations are assumed to be completed during the 5-year readily retrievable period. This ensures that the entire repository formation will have been explored and found to be satisfactory during the readily retrievable period.
- Tests to confirm predicted thermal characteristics of the formation will also be completed during the 5-year readily retrievable period.
- Surface waste handling facilities are designed to accommodate the waste receiving rates shown in Table 7.5.2.

7.5.3 Operations At Reprocessing Fuel Cycle Repositories

Canisters of HLW, FRW, and ILW are received by rail and handled at the repository in the same way as spent fuel canisters (see Section 7.4.3). ILW packaged in 55- and 80-gal drums is shipped in shielded Type B containers by truck to the repository. A crane lifts the containers from the truck bed to shielded transfer cells for remote removal of the drums. The drums are placed in steel drum-pack canisters, each of which holds three drums. Each drum-pack canister is sealed with a welded lid. The drum packs are transported to the canistered waste shaft and lowered into the repository.

Low-level waste arrives at the repository in 55-gal drums and steel boxes. The drums are arranged on pallets in two layers, six drums to a layer. The steel boxes measure 1.2 x 1.8 x 1.8 m (4 x 6 x 6 ft), roughly equivalent in size to the pallet of drums. The LLW is shipped by truck, in Supertiger® cargo carriers. Pallets and boxes are unloaded from the Supertiger® using shielded forklifts. The waste containers are inspected for damage and repaired if necessary, tranported to the low-level waste shaft, and lowered into the repository.

TABLE 7.5.2. Waste Receiving Rates (Salt, Shale)

Year	HLW Canister Diameter (cm)	HLW Canisters	FRW Canisters	ILW Canisters	ILW Drums	LLW Boxes	LLW Drums
	Salt						
Fuel Cycle IIa							
1985	--	0	570	80	9,600	60	3,200
1990	30	700	980	150	17,000	100	5,500
1995	25	1,700	1,300	200	22,000	130	7,400
2000							
2005							
Final Year[(a)]	25	2,300	1,600	240	27,600	170	9,100
Fuel Cycle IIb							
1985	--	0	570	80	9,600	70	3,300
1990	30	700	980	150	17,000	120	5,600
1995	30	1,200	1,300	200	22,000	160	7,600
2000	30	1,800	1,800	270	30,000	220	10,000
2005	30	2,500	2,300	330	38,000	280	13,000
Final Year[(a)]	30	2,500	2,300	330	38,000	280	13,000
Fuel Cycle III							
1985	--	0	570	80	9,600	90	5,500
1990	30	700	980	150	17,000	170	11,000
1995	30	1,700	1,300	200	22,000	230	15,000
2000	25	2,600	1,800	270	30,000	320	20,000
2005							
Final Year[(a)]	25	2,900	2,000	290	33,000	350	23,000
	Shale						
Fuel Cycle IIa							
1985	--	0	570	80	9,600	60	3,200
1990	15	2,900	980	150	17,000	100	5,500
1995	15	4,600	1,300	200	22,000	130	7,400
2000							
2005							
Final Year[(a)]	15	4,600	1,300	200	22,000	140	7,400
Fuel Cycle IIb							
1985	--	0	570	80	9,600	70	3,300
1990	20	1,600	980	150	17,000	120	5,600
1995	15	4,600	1,300	200	22,000	160	7,600
2000							
2005							
Final Year[(a)]	15	4,600	1,300	200	22,000	170	7,500
Fuel Cycle III							
1985	--	0	570	80	9,600	90	5,500
1990	20	1,600	980	150	17,000	170	11,000
1995	15	4,600	1,300	200	22,000	230	15,000
2000							
2005							
Final Year[(a)]	15	4,600	1,300	200	22,000	240	15,600

a. Final Year of Repository Operation:

	Salt	Granite	Shale	Basalt
IIa	1999	2004	1997	2002
IIb	2005	2004	1997	2002
III	2003	2004	1997	2002

TABLE 7.5.2. Waste Receiving Rates (Granite, Basalt) contd

Year	HLW Canister Diameter (cm)	HLW Canisters	FRW Canisters	ILW Canisters	ILW Drums	LLW Boxes	LLW Drums
	Granite						
Fuel Cycle IIa							
1985	--	0	570	80	9,600	60	3,200
1990	20	1,600	980	150	17,000	100	5,500
1995	20	2,600	1,300	200	22,000	130	7,400
2000	20	4,100	1,800	270	24,000	190	10,000
2005							
Final Year[a]	20	5,100	2,100	310	35,700	220	11,800
Fuel Cycle IIb							
1985	--	0	570	80	9,600	70	3,300
1990	20	1,600	980	150	17,000	120	5,600
1995	20	2,600	1,300	200	22,000	160	7,600
2000	20	4,100	1,800	270	30,000	220	10,000
2005							
Final Year[a]	20	5,100	2,100	310	36,000	270	12,000
Fuel Cycle III							
1985	--	0	440	65	7,500	71	4,200
1990	20	1,600	980	150	17,000	170	11,000
1995	20	2,600	1,300	200	22,000	230	15,000
2000	20	4,100	1,800	270	30,000	320	20,000
2005							
Final Year[a]		5,100	2,100	210	36,000	380	24,600
	Basalt						
Fuel Cycle IIa							
1985	--	0	570	80	9,600	60	3,200
1990	20	1,600	980	150	17,000	100	5,500
1995	15	4,600	1,300	200	22,000	130	7,400
2000	15	7,200	1,800	270	24,000	180	10,000
2005							
Final Year[a]	15	7,200	1,800	270	30,000	190	10,000
Fuel Cycle IIb							
1985	--	0	570	80	9,600	70	3,300
1990	20	1,600	980	150	17,000	120	5,600
1995	15	4,600	1,300	200	22,000	160	7,600
2000	15	7,200	1,800	270	30,000	220	10,000
2005							
Final Year[a]	15	7,200	1,800	270	30,500	230	10,300
Fuel Cycle III							
1985	--	0	570	80	9,600	90	5,500
1990	20	1,600	980	150	17,000	170	11,000
1995	15	4,600	1,300	200	22,000	230	15,000
2000	15	7,200	1,800	270	30,000	320	20,000
2005							
Final Year[a]	15	7,200	1,800	270	30,500	330	21,400

a. Final Year of Repository Operation:

	Salt	Granite	Shale	Basalt
IIa	1999	2004	1997	2002
IIb	2005	2004	1997	2002
III	2003	2004	1997	2002

Wastes are received at subsurface transfer stations that are integral shaft structures. A shielded transporter is used to remove two HLW canisters or one FRW/ILW canister from the transfer stations and deliver the canisters to an emplacement area. Shielded forklifts are used to remove the pallets and boxes of LLW and load them onto transporters.

At the conceptual repositories in salt and shale formations, HLW canisters are lowered into vertical holes in the emplacement rooms using the same minimum hole spacing described for spent fuel canisters in the once-through fuel cycle repositories and with an allowable thermal density calculated specifically for the reference HLW's characteristics. In these formations, FRW and ILW are also emplaced in holes; however, the minimum hole spacing is increased to 2.3 m (7.5 ft) because of the larger hole diameters for these wastes.

The conceptual repositories in granite and basalt formations emplace HLW the same way as salt and shale repositories. However, FRW and ILW canisters emplaced in repositories in granite or basalt are placed in sleeves in backfilled trenches running the length of the emplacement rooms. Storage racks hold the sleeves upright in a single row prior to backfilling. The racks allow a minimum spacing of 1 m (3.5 ft) center-to-center.

The LLW pallets and boxes are stacked with shielded forklifts two high along the walls of LLW emplacement rooms.

Details of waste spacing, room arrangements and repository area requirements are provided in Table 7.5.3 for the conceptual repositories in salt, granite, shale and basalt formations.

Emplacement holes are drilled with truck mounted bucket drilling rigs at a rate that keeps pace with the canister receiving rate. The trenches in granite and basalt are excavated using conventional drill and blast techniques. Racks containing emplacement sleeves for FRW/ILW canisters are placed in the trenches and the trenches are backfilled with crushed rock. The sleeves extend from the bottom of the trench to the floor of the room as shown in Figure 7.5.1. The crushed backfill provides shielding. Excess crushed rock from drilling and trenching operations is removed by load-haul-dump units and is transported to underground storage silos. The rock in these silos is used for backfill or is removed to the surface.

These conceptual repositories are operated with the same initial readily retrievable period described for the once-through fuel cycle repositories. Steel sleeves and concrete plugs are used for canisters emplaced in holes to protect the canisters during the retrievable period. Emplacement of HLW canisters in sleeved holes is illustrated in Figure 7.5.2. The FRW/ILW canisters emplaced in trenches at the granite and basalt repositories are always placed in sleeves. Low-level waste does not require this additional protection because it is not emplaced in holes.

Canistered wastes are emplaced using a transporter that can carry two canisters of HLW or one canister of FRW or ILW. A transporter mounted hoist is used to remove the concrete plug from the hole or trench sleeve after which the operator positions the transporter cask (containing the waste canister) directly over the hole. With the aid of instrumentation located in the transporter cab, the operator plumbs the canister over the sleeve. The canister hoist lowers the canister into the sleeve. The hoist is then remotely unlatched and withdrawn, the transporter cask raised, and the plug replaced in the sleeve.

TABLE 7.5.3. Repository Area Allocations and Arrangement

Fuel Cycle		Salt	Granite	Shale	Basalt
All cycles					
	Room Size, m[a]				
	HLW	5.5 x 6 x 170	5.5 x 6 x 170	5.5 x 6 x 170	5.5 x 6 x 170
	FRW/ILW	11 x 6 x 1000	5.5 x 6 x 170	8 x 6 x 170	5.5 x 6 x 170
	LLW	11 x 6 x 460	5.5 x 6 x 170	8 x 6 x 170	5.5 x 6 x 170
	Pillar Width, m				
	HLW	18	18	18	16.5
	FRW/ILW/LLW	9	7.6	18	8.2
IIa - Uranium only recycle, Plutonium in HLW					
	Number of Rooms				
	HLW	934	673	572	806
	FRW/ILW	64	1,192	620	1,005
	LLW	4	44	16	37
	Containers/Room[b]				
	HLW	17	73	66	86
	FRW	1,562	145	144	145
	ILW	5,220	435	432	435
	LLW	31,000	4,464	6,624	4,464
	Center to Center Hole Spacing, m				
	HLW	9.5, one row	2.2, one row	2.5, one row	1.8, one row
	FRW	2.6, four rows[c]	1, one row[d]	2.3, two rows[c]	1, one row[d]
	ILW	2.3, four rows[c]	1, one row[d]	2.3, two rows[c]	1, one row[d]
	Total Room and Pillar Area, ha				
	HLW	380	270	230	290
	FRW	20	35	40	36
	ILW	110	200	240	210
	LLW	4	9	7	9
	Total Emplacement Area, ha				
	HLW	500	360	300	380
	FRW	27	46	53	47
	ILW	140	270	31	270
	LLW	5	12	9	12
	Total Repository Area[e] per 1000 MTHM Equivalent ha				
	HLW	13 - 18[f]	4.1 - 11[f]	7.2 - 13[f]	4.7 - 8.4[f]
	FRW	0.45	0.50	1.1	0.60
	ILW	2.4	2.9	6.5	3.3
	LLW	0.08	0.12	0.19	0.14
	Total	15.9 - 20.9	7.6 - 14.5	15.0 - 20.8	8.7 - 12.4

TABLE 7.5.3. contd

Fuel Cycle	Salt	Granite	Shale	Basalt
IIb - Uranium only recycle, Plutonium stored				
Number of Rooms				
HLW	794	668	555	714
FRW/ILW	105	1,190	626	1,013
LLW	7	46	16	40
Containers/Room[b]				
HLW	31	73	66	86
FRW	1,562	145	144	145
ILW	5,220	435	532	435
LLW	31,000	4,464	6,624	4,464
Center to Center Hole Spacing, m				
HLW	5.3, one row	2.2, one row	2.5, one row	1.8, one row
FRW	2.6, four rows[c]	1, one row[d]	2.3, two rows[c]	1, one row[d]
ILW	2.3, four rows[c]	1, one row[d]	2.3, two rows[c]	1, one row[d]
Total Room and Pillar Area, ha				
HLW	320	270	220	260
LLW	36	35	40	36
ILW	170	205	240	210
LLW	6	9	7	10
Total Emplacement Area, ha				
HLW	420	350	290	340
FRW	47	46	53	47
ILW	230	270	310	280
LLW	8	12	10	13
Total Repository Area[e] per 1000 MTHM Equivalent ha				
HLW	6.6 - 9.3[f]	4.1 - 6.2[f]	7.2 - 13[f]	3.1 - 8.4[f]
FRW	0.45	0.50	1.1	0.60
ILW	2.4	2.9	6.5	3.5
LLW	0.08	0.12	0.19	0.14
Total	9.5 - 12.2	7.6 - 9.7	15.0 - 20.8	7.3 - 12.6
III - Uranium and Plutonium Recycle				
Number of Rooms				
HLW	830	668	555	738
FRW/ILW	95	1,190	626	1,013
LLW	10	84	29	72

TABLE 7.5.3. contd

Fuel Cycle		Salt	Granite	Shale	Basalt
III - Uranium and Plutonium Recycle (contd)					
	Containers/Room(b)				
	HLW	31	73	66	86
	FRW	1,562	145	144	145
	ILW	5,220	435	432	435
	LLW	31,000	4,464	6,624	4,464
	Center to Center Hole Spacing, m				
	HLW	5.3, one row	2.2, one row	2.5, one row	1.8, one row
	FRW	2.6, four rows(c)	1, one row(d)	2.3, two rows(c)	1, one row(d)
	ILW	2.3, four rows(c)	1, one row(d)	2.3, two rows(c)	1, one row(d)
	Total Room and Pillar Area ha				
	HLW	340	270	220	270
	FRW	30	35	40	36
	ILW	160	210	240	210
	LLW	10	17	13	17
	Total Emplacement Area ha				
	HLW	440	350	290	350
	FRW	40	46	50	47
	ILW	210	270	310	280
	LLW	13	22	17	23
	Total Repository Area(e) per 1000 MTHM Equivalent ha				
	HLW	6.6 - 15(f)	4.1 - 11(f)	7.2 - 13(f)	3.1 - 8.4(f)
	FRW	0.45	0.50	1.1	0.60
	ILW	2.4	2.9	6.5	3.5
	LLW	0.08	0.12	0.19	0.14
	Total	9.5 - 17.9	7.6 - 14.5	15.0 - 20.8	7.3 - 12.6

a. Width by height by length.
b. Canisters are not emplaced in the first 10 m of the rooms.
c. Multiple rows of holes are spaced 1.8 m center to center between the rows.
d. In granite and basalt formations, FRW and ILW are placed into trenches 1.7 m (5.6 ft) wide.
e. Amount of total repository area, including shaft, maintenance, corridor and emplacement areas required for emplacement of 1000 MTHM equivalent waste. Total area for a single repository (as conceptualized in this report) is approximately 800 ha (2000 acres).
f. As canister diameter changes to meet the kW/canister constraint (see Section 7.3), the number of MTHM equivalent per canister also changes. For the constant canister spacing assumed in these conception designs the amount of repository area per MTHM equivalent changes inversely with the canister size. The range of canister sizes used at the repositories are:

Canister Diameter, cm

Fuel Cycle	Salt	Granite	Shale	Basalt
IIa	25-30	15-25	15-20	15-20
IIb	25-30	20-25	15-20	15-25
III	20-30	15-25	15-20	15-25

The 30 cm (12 in.) diameter canister contains 3 MTHM equivalent waste while the 25, 20, and and 15 cm diameter canisters contain 2.1, 1.3, and 0.77 MTHM equivalent waste respectively.

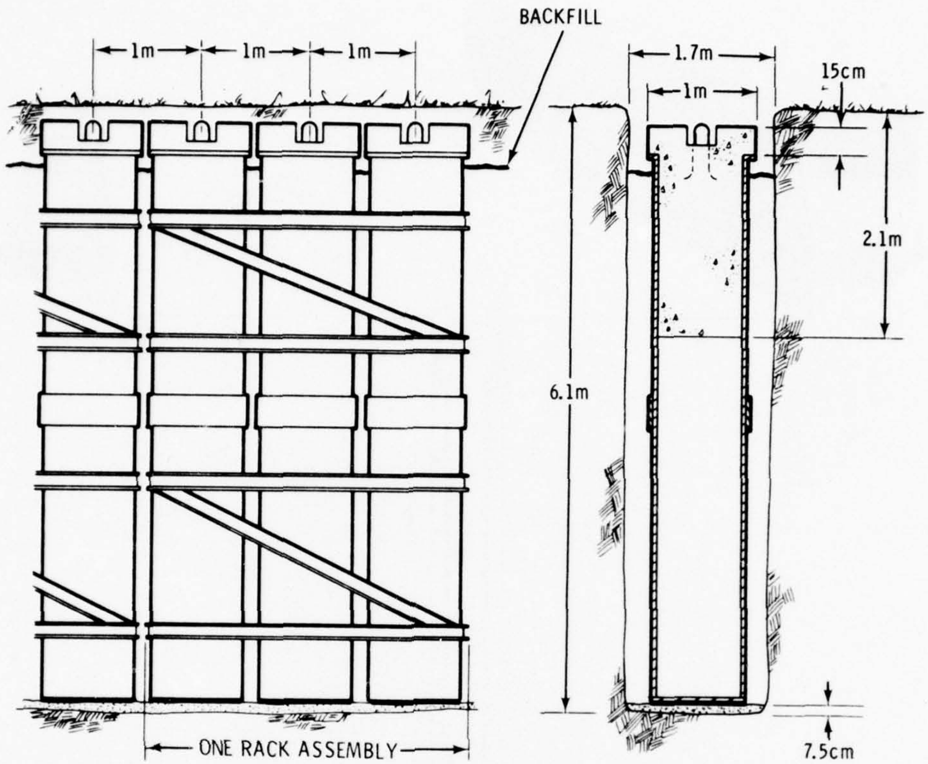


FIGURE 7.5.1. FRW/ILW Emplacement Trenches - Reprocessing Fuel Cycle Repositories in Granite and Basalt

Shielded forklifts remove the palletized LLW drums and the LLW boxes from transporters after the waste arrives at the LLW emplacement area. The forklifts stack the drums and boxes along each side of the LLW room leaving a central aisleway for continued access.

After the readily retrievable period, use of sleeves in emplacement holes is discontinued and the holes are backfilled with crushed rock after canister placement. After rooms become filled to capacity with canisters they are backfilled to within 0.6 m (2 ft) of the ceiling with crushed rock from prior repository mining. Table 7.5.4 lists the contents at the end of emplacement for alternative conceptual first repositories located in salt, granite, shale and basalt formations.

7.5.4 Facility Description for Reprocessing Fuel Cycle Repositories

The conceptual repositories consist of surface facilities for waste receiving and handling, mine support, and mined rock storage, and subsurface facilities for waste handling and emplacement, and mined rock removal.

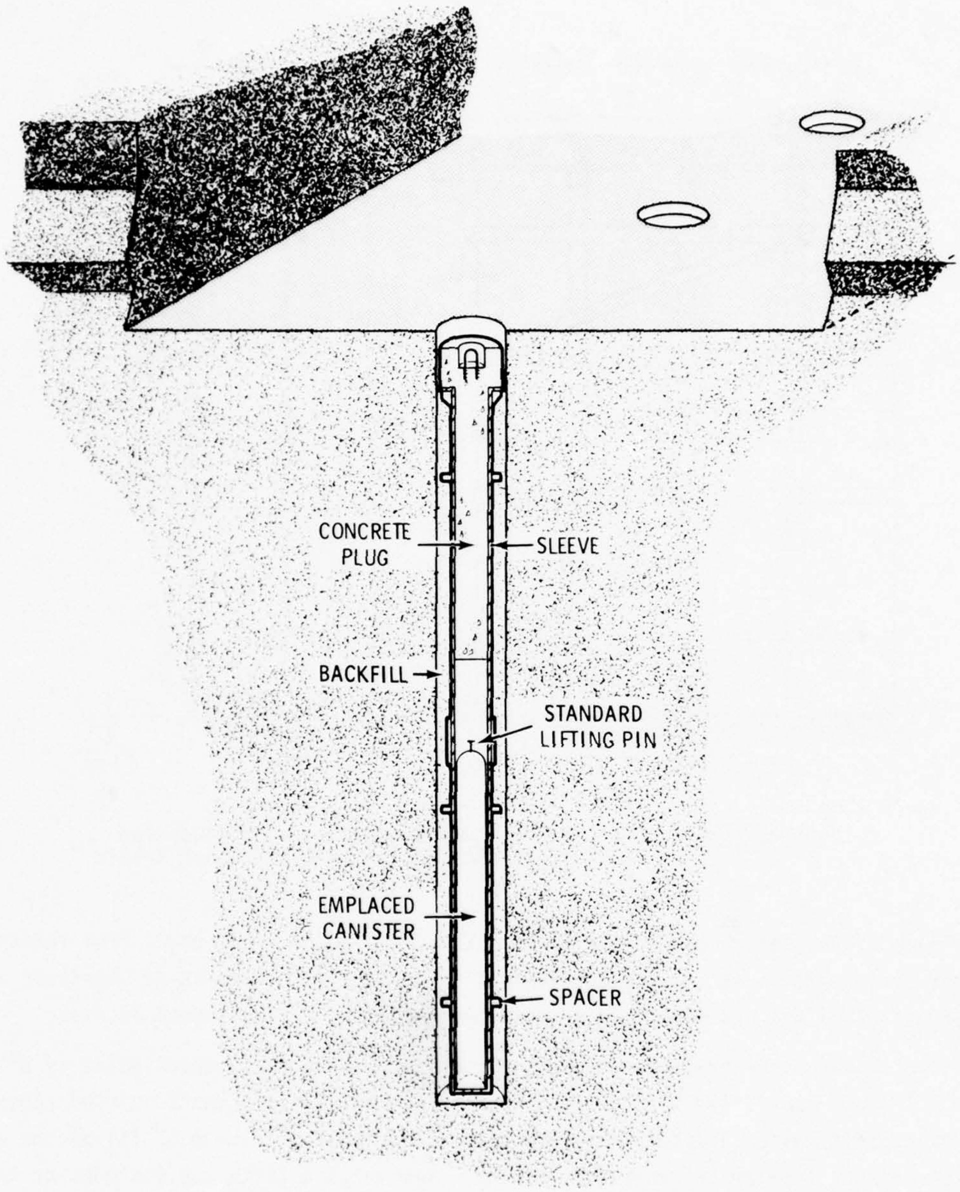


FIGURE 7.5.2. Emplacement of HLW Canisters in Sleeved Holes

The surface facilities shown in Figure 7.5.3, the facility plot plan, are the only visible evidence of the repository. These facilities occupy an area of 180 ha (440 acres) at the salt and shale repositories and 220 ha (540 acres) at the granite and basalt repositories. The additional 100 ha at the granite and basalt repositories are required for surface storage of the larger amounts of rock that are mined from these formations.

The overall subsurface areas consisting of service areas, corridors and waste emplacement rooms occupy 800 ha in all cases.

TABLE 7.5.4. Contents of Alternative First Repositories

Fuel Cycle	Waste	Salt Containers	Salt Equivalent(a) MTHM	Granite Containers	Granite Equivalent MTHM	Shale Containers	Shale Equivalent MTHM	Basalt Containers	Basalt Equivalent MTHM
IIa - Uranium only recycle, Plutonium in HLW									
		(1999)[c]		(2004)		(1997)		(2002)	
	HLW canisters[b]	15,900	39,500	49,200	69,000	37,700	30,500	69,600	56,000
	FRW canisters	15,900	69,000	25,300	108,500	13,100	56,000	21,500	91,500
	ILW canisters	2,370	69,000	3,760	108,500	1,950	56,000	3,190	91,500
	ILW drums	270,000	69,000	427,000	108,500	222,000	56,000	363,000	91,500
	LLW boxes	1,670	69,000	2,650	108,500	1,370	56,000	2,250	91,500
	LLW drums	88,500	69,000	141,000	108,500	72,800	56,000	119,000	91,500
IIb - Uranium only recycle, Plutonium stored									
		(2005)		(2004)		(1997)		(2002)	
	HLW canisters[b]	24,600	76,500	48,800	69,000	36,600	30,500	61,700	56,000
	FRW canisters	27,500	118,000	25,300	108,500	13,100	56,000	21,500	91,500
	ILW canisters	4,090	118,000	3,760	108,500	1,950	56,000	3,190	91,500
	ILW drums	469,000	118,000	431,000	108,500	224,000	56,000	367,000	91,500
	LLW boxes	3,450	118,000	3,170	108,500	2,290	56,000	2,700	91,500
	LLW drums	158,000	118,000	145,000	108,500	75,500	56,000	124,000	91,500
III - Uranium and Plutonium recycle									
		(2003)		(2004)		(1997)		(2002)	
	HLW canisters[b]	25,800	62,170	48,800	69,000	36,600	30,500	63,800	56,000
	FRW canisters	23,400	99,670	25,300	108,500	13,100	56,000	21,500	91,500
	ILW canisters	3,480	99,670	3,760	108,500	1,950	56,000	3,190	91,500
	ILW drums	399,000	99,670	431,000	108,500	224,000	56,000	367,000	91,500
	LLW boxes	4,150	99,670	4,500	108,500	2,290	56,000	3,810	91,500
	LLW drums	264,000	99,670	286,000	108,500	144,000	56,000	242,000	91,500

a. Tonnes of reprocessed heavy metal corresponding to wastes contained in the repository.
b. Equivalent MTHM for HLW does not correspond to the MTHM for other waste types because of 5 year delay for HLW cooling between reprocessing and shipment to the repository.
c. Final year of operation assuming 1985 repository startup.

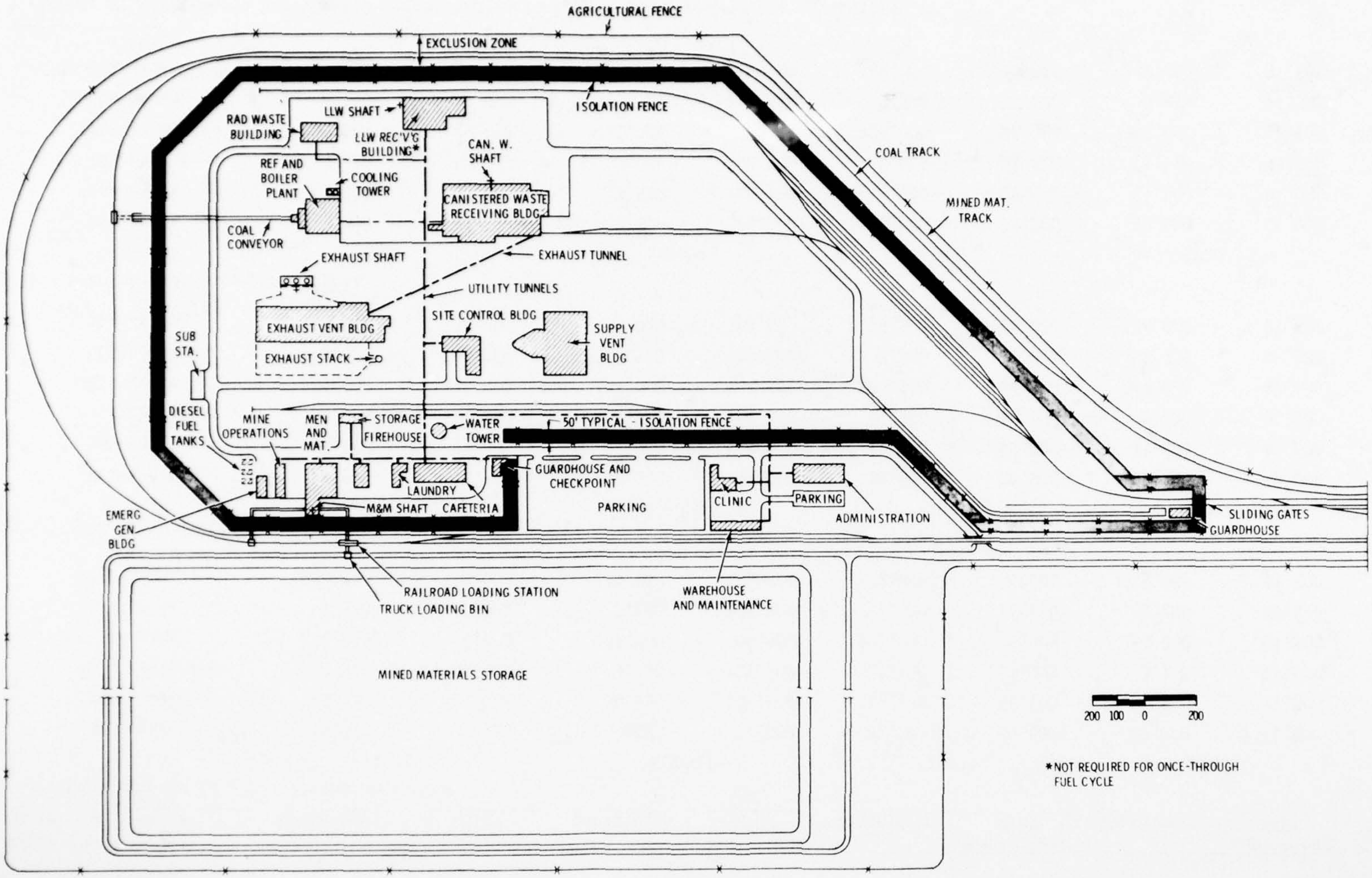


FIGURE 7.5.3. Facility Plot Plan - Reprocessing Fuel Cycle Repositories

7.5.4.1 Surface Facilities

All surface facilities are surrounded by an agricultural fence as shown in Figure 7.5.3. An additional double security fence is provided for the waste handling facilities and repository shafts. Major facilities within the security fence include:

- canistered waste building
- low-level waste building
- radioactive waste treatment building
- exhaust ventilation building
- supply ventilation building
- men and material building

Repository surface facilities that are of Category I* design are the canistered waste building, low-level waste building, radioactive waste treatment building, exhaust ventilation building, supply ventilation building, and standby generator building. The various equipment and parts incorporated into the repository design are assumed to be, to the greatest extent possible, off-the-shelf items of standard manufacture.

Canistered Waste Building. The canistered waste (CW) building has facilities and equipment for receiving and handling radioactive wastes that require remote handling and a high degree of attenuation shielding when removed from their shielded shipping casks. The CW building is a two-story reinforced concrete structure (one story above grade and one below grade). The building is designed according to criteria specified in 10 CFR Part 50, Appendix A.

All areas where the wastes are outside of a shielded cask have adequate biological shielding of conventional reinforced concrete. All activities in these areas (transfer cells, canister feed room, waste transfer gallery, and shaft transfer gallery) are remotely monitored and controlled.

Figures 7.5.4 through 7.5.7 provide plan and section views of the CW building for a repository in salt operating in fuel cycle IIb or III. The ground floor contains four receiving and cask preparation bays and nine transfer cells as shown in Figure 7.5.4. Fuel cycle IIa in salt requires only three receiving bays and seven transfer cells. One of the bays in each case is assigned to receive the canistered wastes that arrive by rail. This bay feeds into three cask transfer galleries located in the basement (Figure 7.5.5). These galleries feed into transfer cells and overpack facilities that process the two types of canisters (HLW and FRW/ILW) that arrive at this bay. The remaining bays receive drummed wastes that arrive by truck. Each of these bays feeds two lines of transfer galleries, transfer cells, and overpack facilities. Also located in the basement of this building are two shift feed galleries.

As shown in Table 7.5.2, repositories located in the other geologic media receive wastes at different rates than those for the salt facility. As a result, the waste handling capabilities of the CW building are modified at the different repositories by adding or deleting waste

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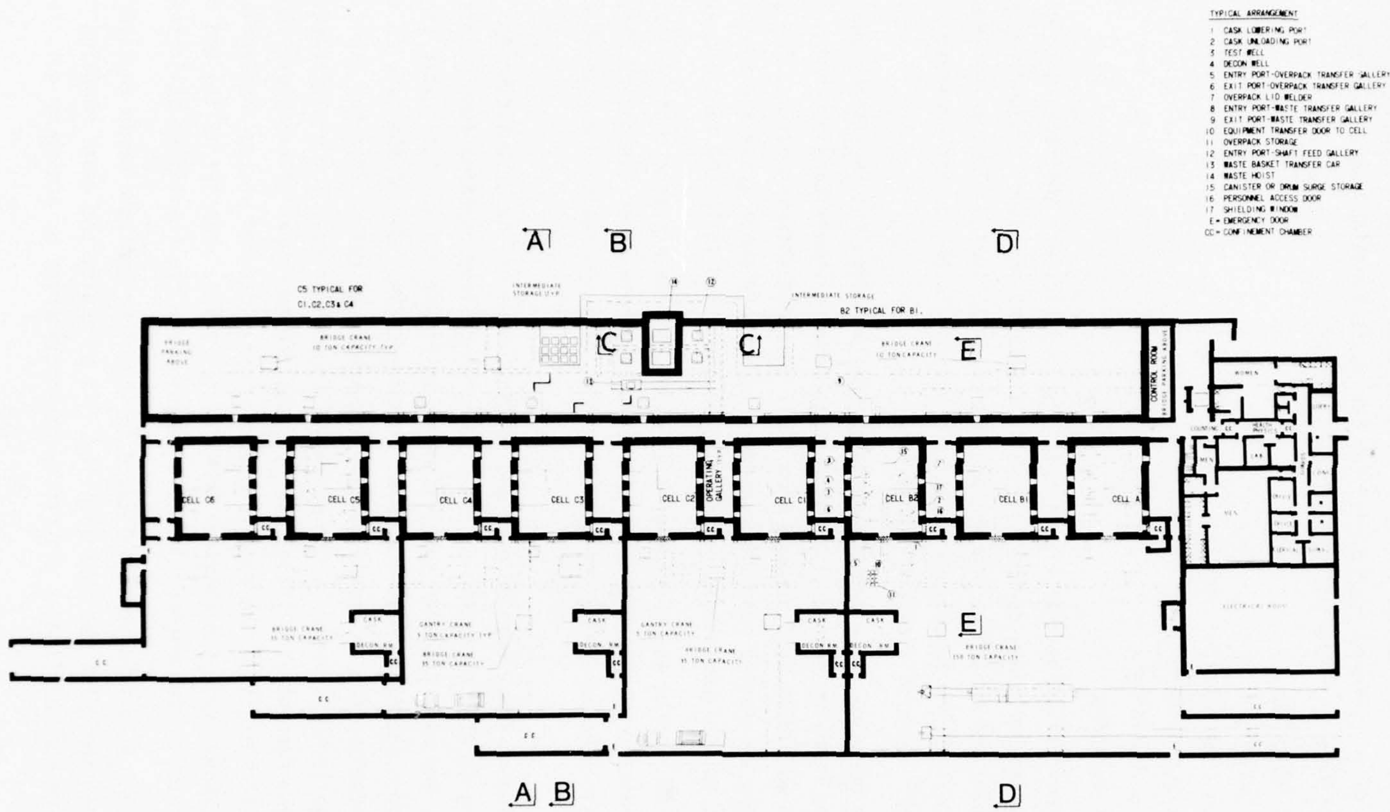
* Designed to withstand maximum credible natural disasters, such as earthquakes and tornadoes.



<u>FIGURE 7.5.4.</u> Canistered Waste Building Ground Floor Plan, Reprocessing Fuel Cycles IIb and III Repositories in Salt

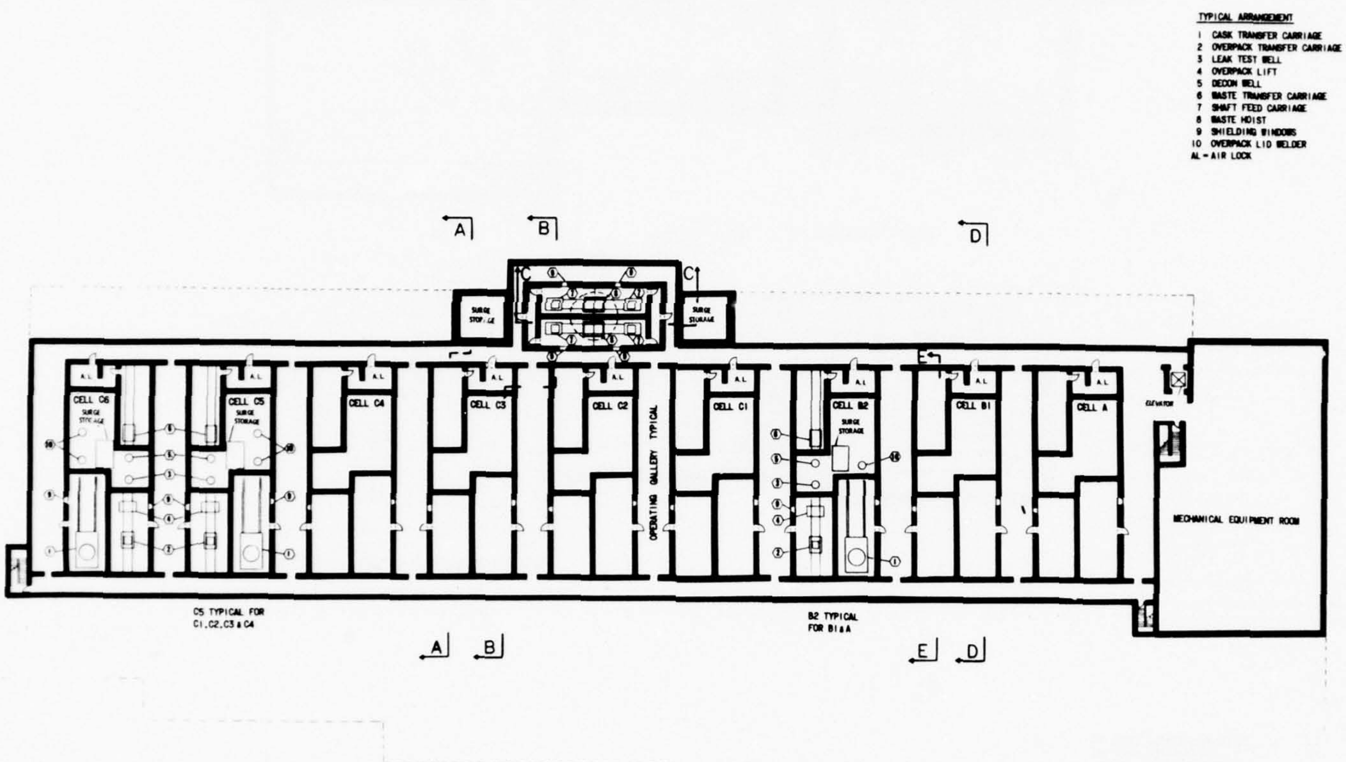


FIGURE 7.5.5. Canistered Waste Building Basement Plan, Reprocessing Fuel Cycles IIb and III Repositories in Salt

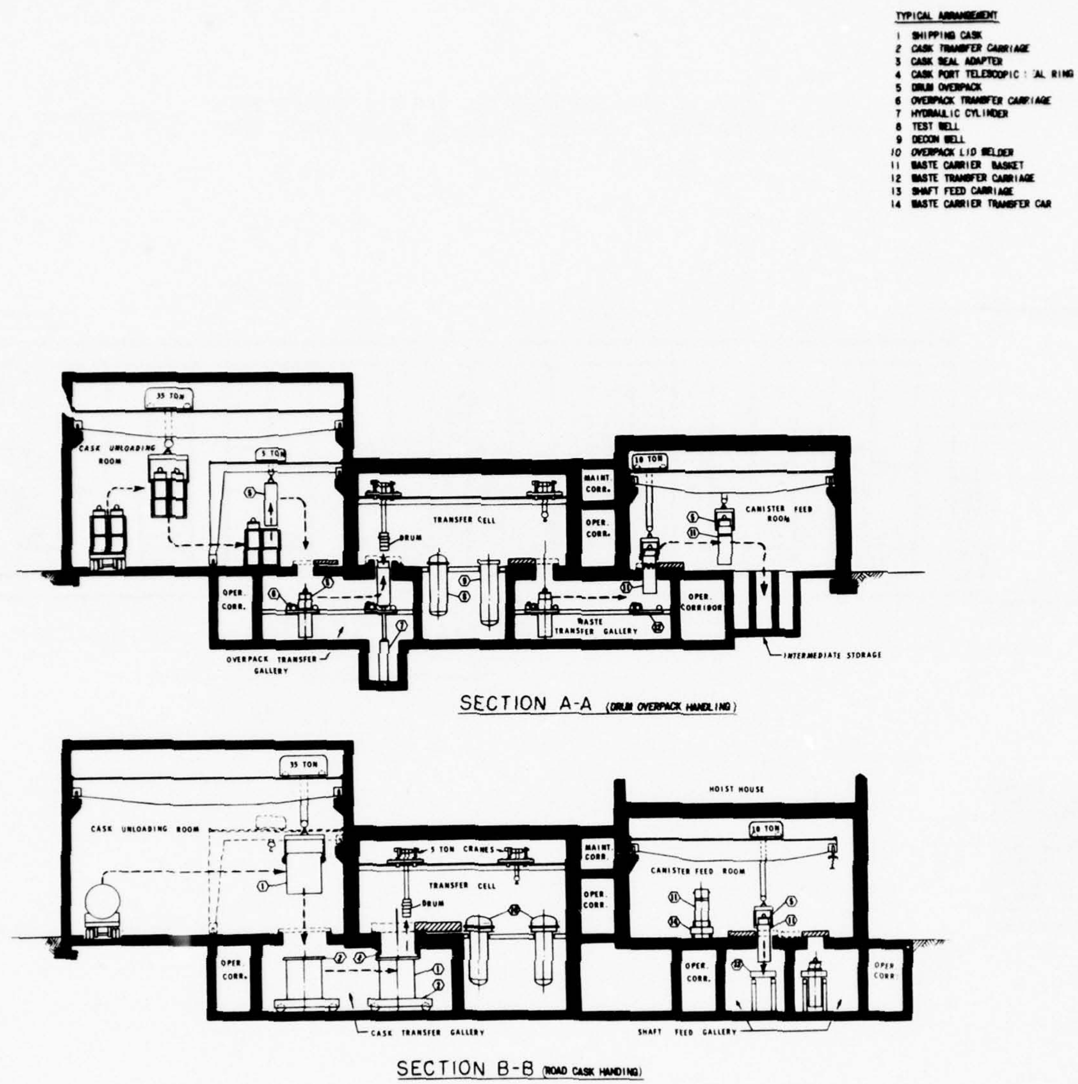


FIGURE 7.5.6. Canistered Waste Building Section Views, Reprocessing Fuel Cycles IIb and III Repositories in Salt

TYPICAL ARRANGEMENT

1 SHIPPING CASK
2 CASK TRANSFER CARRIAGE
3 CASK SEAL ADAPTER
4 CASK PORT TELESCOPIC SEAL RING
5 CANISTER OVERPACK
6 OVERPACK TRANSFER CARRIAGE
7 HYDRAULIC CYLINDER
8 TEST WELL
9 DECON WELL
10 OVERPACK LID WELDER
11 WASTE CARRIER BASKET
12 WASTE TRANSFER CARRIAGE
13 SHAFT FEED CARRIAGE
14 RAIL CASK TRANSPORTER
15 CLADDING WASTE CANISTER
16 HLW CANISTER

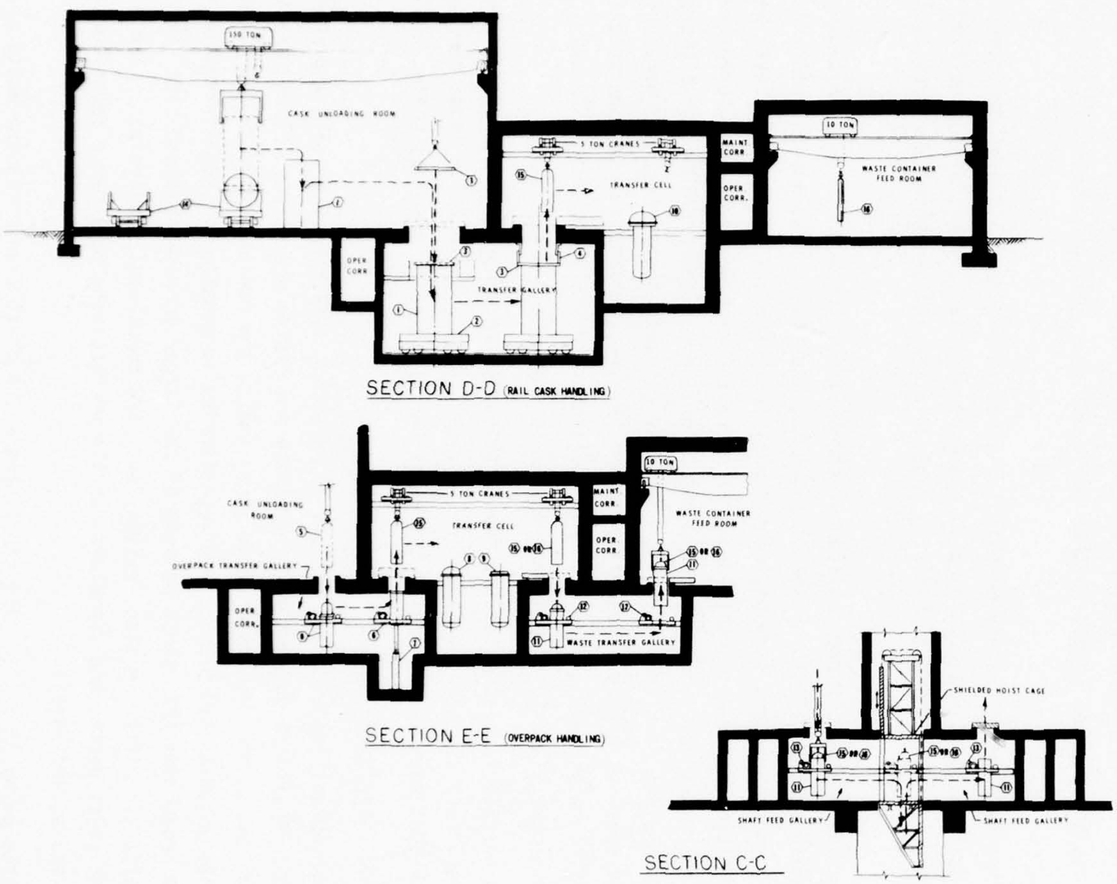


FIGURE 7.5.7. Canistered Waste Building Section Views, Reprocessing Fuel Cycles IIb and III Repositories in Salt

handling equipment and cells. At the repositories in granite and basalt a complete handling module (transfer gallery, transfer cell, and overpack station) is added. At the shale facility, one module is deleted from the CW building.

The processed waste containers from all lines are transported by overhead bridge cranes to the shaft feed gallery beneath the waste container feed room.

Additional space in this building is provided for support functions and activities including electrical and mechanical rooms, health physics room, laboratory, and other personnel and maintenance facilities.

Low-level Waste Building. The low-level waste receiving building has facilities and equipment for receiving and handling waste materials that can be contact handled or that require less attenuation shielding than waste assigned to the canistered waste receiving building.

The low-level waste receiving building shown in Figure 7.5.8 is a one story reinforced concrete building designed according to the general intent of criteria specified in 10 CFR Part 50, Appendix A. The same building configuration and arrangement serves Cycles IIa, IIb, and III.

The building provides two confinement chambers for incoming trucks, unloading docks, an area for maneuvering pallets by forklift trucks, two test cells, one decontamination and overpack cell, empty overpack storage, confinement chamber access to the Low-Level Waste Shaft, a battery charging room, and a shielded area where incoming waste may be stored temporarily whenever the waste hoist is shut down. Additional areas are provided for a mechanical equipment room (housing fans and filters), an electrical switchgear room, and personnel facilities. A railroad spur has been provided into one of the truck unloading stations to allow receipt of LLW via railcars as well as trucks. The confinement chamber for access to this unloading station has been sized to accommodate a rail car and switcher.

Other surface facilities at the repositories for reprocessing fuel cycles are the same as once-through fuel cycle facilities (see Section 7.4.4). Table 7.5.5 lists the quantities of mined rock that must be managed at the repositories in salt, granite, shale, and basalt.

7.5.4.2 Shafts and Hoists

The conceptual repositories for the reprocessing fuel cycles in salt and shale formations require four shafts to support waste handling and mining operations. These are the canistered waste (CW) shaft, the low-level waste (LLW) shaft, the men and material (M&M) shaft and the ventilation exhaust (VE) shaft. The repositories in granite and basalt require the addition of a mine production (MP) shaft because of the larger amounts of mined rock (see Table 7.5.5). These shafts all differ in size, design, use, and functional constraints. Table 7.5.6 summarizes shaft depths and diameters for the reprocessing fuel cycle repositories in salt, granite, shale, and basalt.

Canistered Waste Shaft. The canistered waste shaft transports the waste canisters from the canistered waste building to the subsurface emplacement areas. It is identical to the CW shaft described in Section 7.4.4 (Figure 7.4.9) for the once-through fuel cycle repositories.

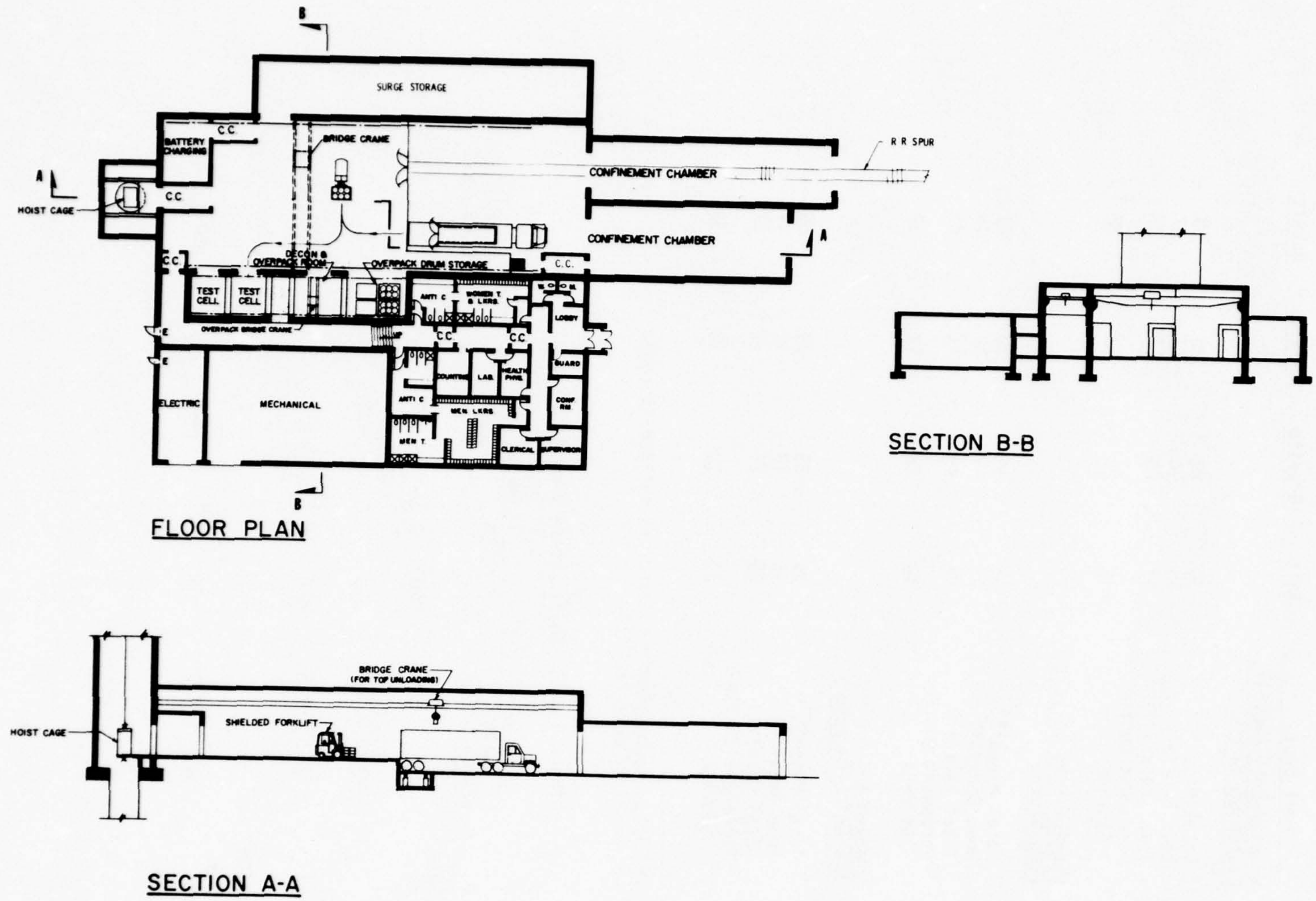


FIGURE 7.5.8. LLW Building - Plan and Sections, Reprocessing Fuel Cycle Repositories

TABLE 7.5.5. Mining and Rock Handling Requirements (MT x 10^6)

Fuel Cycle	Salt	Granite	Shale	Basalt
Uranium-Only Recycle, Plutonium in HLW				
Total Mined[a]	30	52	30	59
Room Backfill	13	16	12	17
Total Backfill[b]	17	25	17	27
Permanent Onsite Storage	13	27	13	32
Uranium-Only Recycle, PuO_2 Stored				
Total Mined[a]	36	52	30	57
Room Backfill	15	17	12	16
Total Backfill	20	24	17	25
Permanent Onsite Storage	16	28	13	32
Uranium and Plutonium Recycle				
Total Mined[a]	35	53	30	59
Room Backfill	15	17	12	17
Total Backfill	20	24	17	27
Permanent Onsite Storage	15	29	13	32

a. This amount requires temporary surface storage.
b. Includes room backfill.

TABLE 7.5.6. Shaft Depths and Diameters, m

	Media							
	Salt		Granite		Shale		Basalt	
Shaft	depth	dia	depth	dia	depth	dia	depth	dia
Canistered Waste	580	4.3	620	4.3	460	4.3	600	4.3
Low-Level Waste	610	3.0	610	3.0	490	3.0	610	3.0
Men and Materials	580	9.5	670	8.2	510	9.5	640	8.2
Mine Production	--	--	670	8.2	--	--	640	8.2
Ventilation Exhaust	560	7.9	610	8.5	440	7.9	580	8.5

At mine level, the shaft provides access to the FRW/ILW receiving station at one elevation and the HLW receiving station 23 m (75 ft) lower. The location of the FRW/ILW and HLW receiving stations and their respective mine areas on different elevations is required to provide space for routing men and materials corridors to each mine area. The receiving stations provide shielded facilities for remote transfer of canisters from the shaft.

Low-Level Waste Shaft. The low-level waste shaft transports LLW pallets and boxes from the low-level waste building on the surface to the LLW subterranean emplacement areas. This shaft has a 3-m (10-ft) inside diameter and is lined with a minimum of 0.3 m (1 ft) of concrete. The shaft connects the LLW building with the subterranean receiving station, where pallets and boxes are picked up by shielded forklifts and loaded onto transporters.

Men and Materials Shaft. Men and materials (M&M) shafts at the reprocessing fuel cycle repositories are identical in function and layout to the M&M shafts described in Section 7.4.4 and Figures 7.4.10 and 7.4.11 for the once-through fuel cycle repository. Depths and diameters for the reprocessing fuel cycle M&M shafts are shown in Table 7.5.6.

Ventilation Exhaust Shaft. Ventilation air from the mine areas is exhausted through this shaft into filtration systems located on the surface. The shaft is divided into two compartments to provide separate exhaust for the mining and emplacement operations.

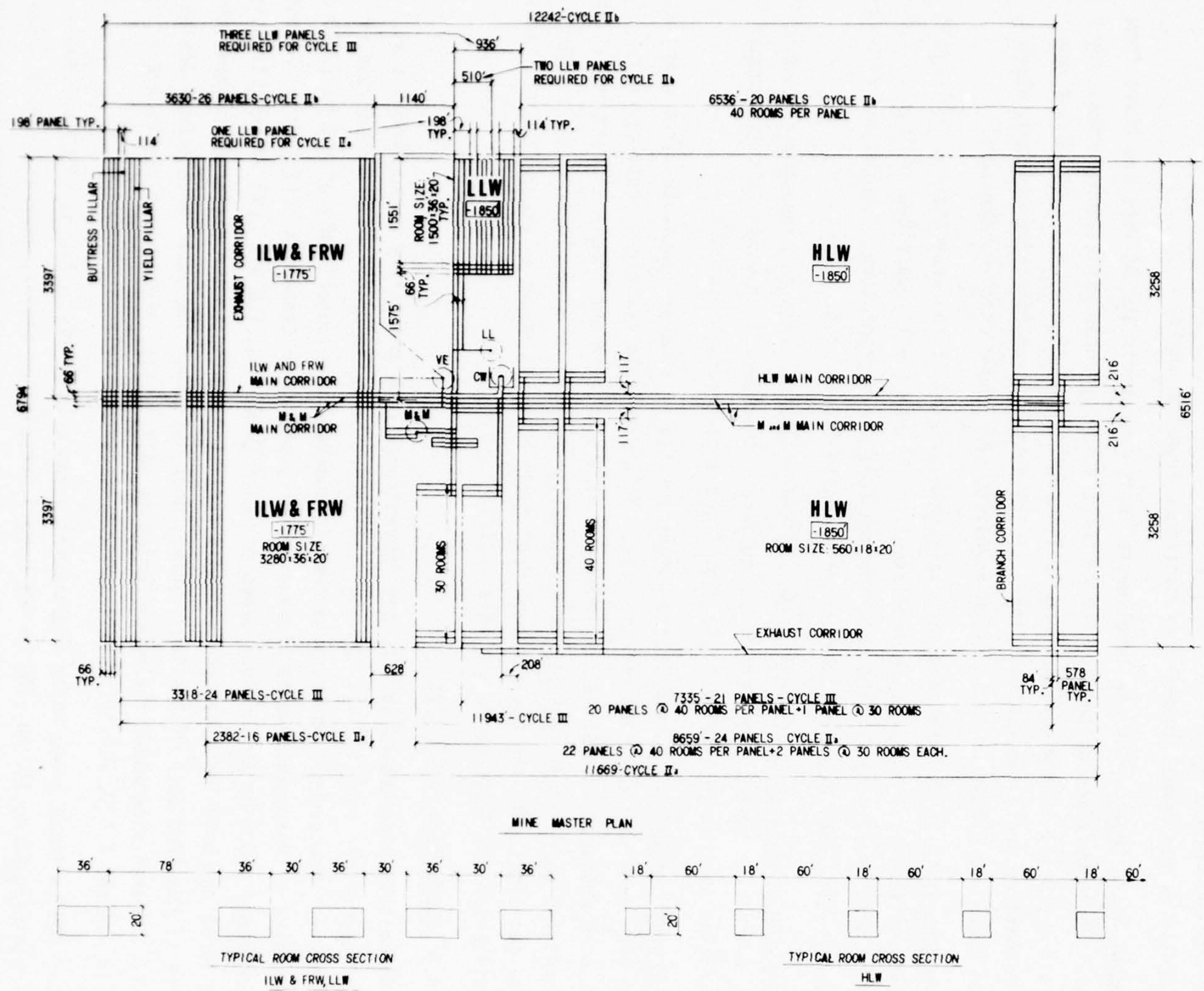
Mine Production Shaft. Substantially larger amounts of rock are excavated from granite and basalt than from salt or shale repositories. To accommodate the higher rate of rock removal, a mine production (MP) shaft is provided at the granite and basalt repositories. The MP shaft contains three 23-MT capacity skip hoists for removal of mined rock to the surface and is laid out similar to the once-through fuel cycle MP shaft shown in Figure 7.4.12. Additional ventilation air is supplied via this shaft.

7.5.4.3 Subsurface Facilities

The repository underground layouts are conventional room and pillar arrangements that provide for repository ventilation, opening stability, thermal effects and efficient use of excavated space. The overall underground area is bounded by an upper limit of 800 ha (2000 acres) set on the basis of reasonable waste storage capacity and waste transport efficiency. Allocation of the repository's 800 ha to the four types of waste (HLW, FRW, ILW and LLW) varies slightly among the three reprocessing fuel cycles. This variation, owing to the different thermal characteristics of HLW from each of the fuel cycles, does not alter the basic underground design and layout. General mine plans for repositories in salt, shale, granite and basalt are provided in Figures 7.5.9, 7.5.10, 7.5.11 and 7.5.12 respectively.

Of the 800-ha total area, waste emplacement areas occupy 650 to 730 ha (1600 to 1800 acres) with the remaining 80 to 160 ha (200 to 400 acres) occupied by shafts, general service areas, main corridors and unmined areas within the repository. A summary of repository area allocations and arrangement is provided in Table 7.5.3.

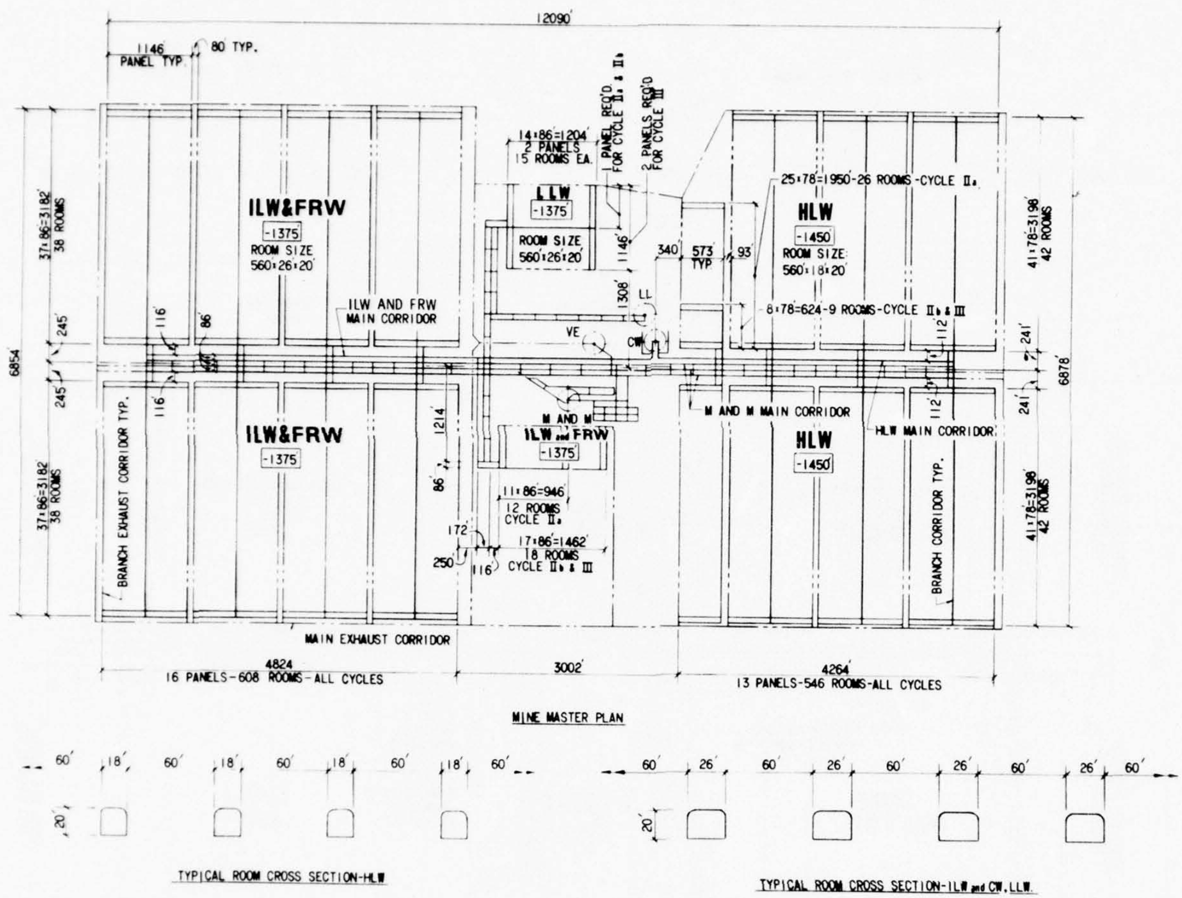
Located in the central shaft area at each repository are support facilities for mining and emplacement activities. These include a communications center, a warehouse and mechanical/electrical shops, parking areas, and equipment assembly areas. These facilities are located outside the 46-m (150-ft) radius safety zone provided around each shaft.



NOTES: 1. ALL LINES REPRESENT CENTER LINES OF ROOMS OR CORRIDORS

2. ALL CORRIDORS ARE 30' WIDE X 20' HIGH

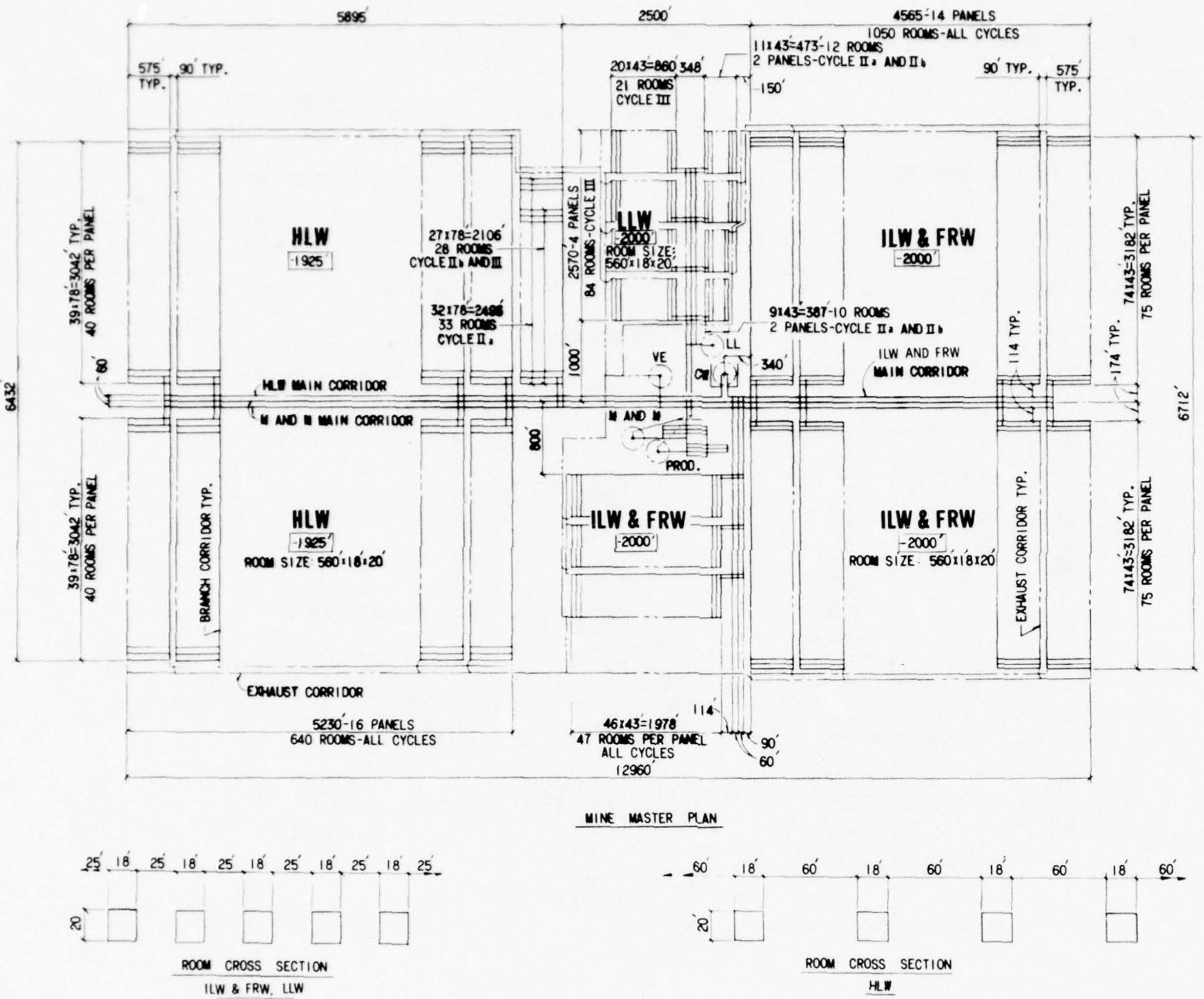
FIGURE 7.5.9. Mine Master Plan - Reprocessing Fuel Cycle Repository in Salt



NOTES:

1. ALL LINES REPRESENT CENTER LINES OF ROOMS OR CORRIDORS.
2. ALL CORRIDORS ARE 26' WIDE x 20' HIGH

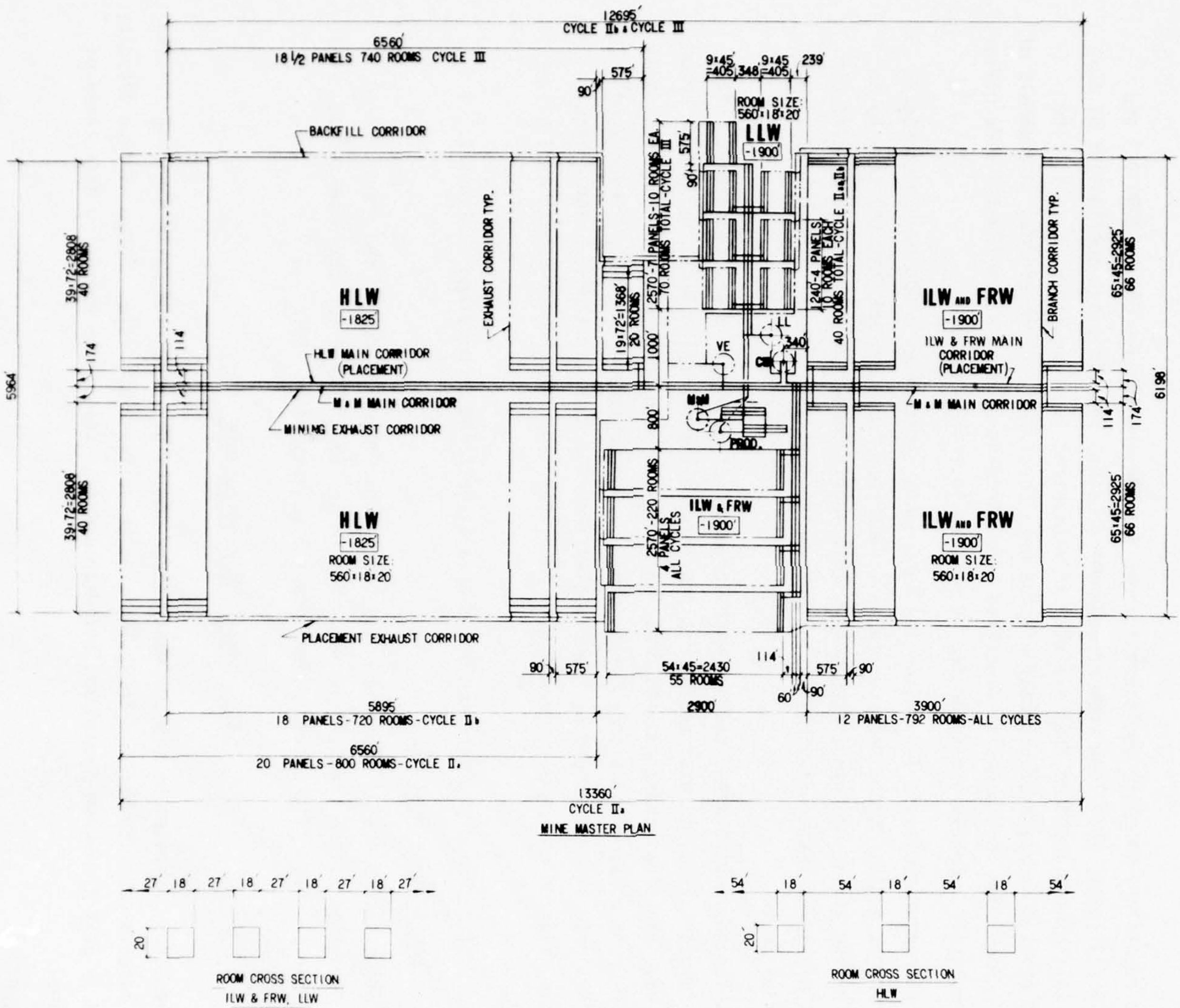
FIGURE 7.5.10. Mine Master Plan - Reprocessing Fuel Cycle Repository in Shale



NOTES:

1. ALL LINES REPRESENT CENTER LINES OF ROOMS OR CORRIDORS
2. ALL CORRIDORS ARE 30' WIDE X 20' HIGH

FIGURE 7.5.11. Mine Master Plan - Reprocessing Fuel Cycle Repository in Granite



NOTES: 1. ALL LINES REPRESENT CENTER LINES OF ROOMS OR CORRIDORS

2. ALL CORRIDORS ARE 30' WIDE X 20' HIGH

FIGURE 7.5.12. Mine Master Plan - Reprocessing Fuel Cycle Repository in Basalt

Extending out from the shafts are the men and materials (M&M) and waste transport main corridors. The M&M corridors provide access for men, equipment, and mined rock between the shafts and emplacement areas. Waste handling operations use the waste transport corridors. These corridors also carry ventilation supply air from the shafts to the appropriate mine areas.

At the repository in salt, emplacement rooms for FRW and ILW are 610 m (2000 ft) and 1000 m (3300 ft) long. This long room arrangement provides for efficient application of the continuous salt mining machines that are used in excavating the repository in salt. The emplacement rooms are excavated directly off the main corridors and are grouped into panels of four rooms. Each room in a panel is separated from the next room by a 9-m (30-ft) wide pillar of intact rock salt. Panels are separated by pillars 24 m (78 ft) wide.

HLW emplacement rooms are 170 m (560 ft) long and are excavated along branch corridors that are at right angles to the main corridors. Each room is separated from the next by an 18-m (60-ft) wide pillar of intact rock salt. This short room arrangement is more difficult to mine but provides greater flexibility for HLW emplacement and increased control over the dissipation of heat.

Non-salt repositories are excavated with conventional drill and blast techniques. The short room layout (170 m) is used because it provides multiple working faces for mining crews. Rubble removal, blasting, charging, and drilling activities can be carried out simultaneously by different crews working in different room panels. The emplacement rooms are excavated along branch corridors that extend at right angles from the main corridors. Each branch corridor is considered one panel of rooms with each room separated from the next by a pillar of intact rock.

A summary of repository arrangements and area allocations is provided in Table 7.5.3.

7.5.4.4 Ventilation Systems

Ventilation system design criteria and operating characteristics for the reprocessing fuel cycle repositories are the same as described in Section 7.4.4.4 for the once-through fuel cycle repositories. Specific system parameters are summarized in Table 7.5.7.

7.5.5 Operating Requirements

The repository operates on a 5-day week, three shifts per day. The waste handling facilities allow reserve capacity for any irregularities in delivery schedule. Surge storage capacity is provided in both the above-ground and below-ground facilities. An operating efficiency of 67% is assumed.

Manpower

The operating labor force at the repositories includes a variety of craft, engineering, management, and miscellaneous support personnel involved in the following major activities:

TABLE 7.5.7. Mine Ventilation Summary

	Salt	Granite	Shale	Basalt
Ventilation Supply				
Fresh air required, m^3/min[a]				
Mining operations	11,000	17,000	11,000	19,000
Hole drilling	7,800	7,800	7,800	7,800
Miscellaneous areas	9,900	15,000	9,900	17,000
20% Recirculation	5,800	7,900	5,800	9,900
Waste Emplacement	7,100	7,100	7,100	7,100
Total	42,000	55,000	42,000	61,000
Number of fans, operating/backup	6/2	8/2	6/2	9/1
Fan capacity, m^3/min	6,800	7,100	6,800	7,100
Ventilation Exhaust				
Mining[a]				
m^3/min	34,000	51,000	34,000	51,000
Number of fans, operating/backup	2/1	3/1	2/1	3/1
Fan capacity, m^3/min	17,000	17,000	17,000	17,000
Emplacement[b]				
m^3/min	7,100	7,100	7,100	7,100
Number of fans, operating/backup	1/1	1/1	1/1	1/1
Fan capacity, m^3/min	7,100	7,100	7,100	7,100

a. Air is filtered through a 30% roughing filter.
b. Air is filtered through a 90% roughing filter followed by two HEPA filters in series.

- waste handling
- hole drilling and trenching
- sleeve placement
- back filling
- maintenance.

Support functions for these activities include:

- radiation monitoring
- utilities
- excess mined rock management
- general administration.

Labor requirements at the repositories are primarily dependent on the waste receiving rate and will therefore vary from year to year. Table 7.5.8 lists total labor force requirements by year of operation for the repositories in salt, granite, shale, and basalt. Labor requirements for construction of surface facilities and shafts, and subsurface mining are described in Section 7.5.11.

TABLE 7.5.8. Repository Operating Staff

Year	Salt Operations[a]	Salt Drilling and Backfill[b]	Granite Operations	Granite Drilling and Backfill	Shale Operations	Shale Drilling and Backfill	Basalt Operations	Basalt Drilling and Backfill
	Man Yrs/Yr. Fuel Cycle IIa							
1980	60	--	60	--	60	--	60	--
1981	65	--	65	--	65	--	65	--
1982	80	--	80	--	80	--	80	--
1983	100	--	100	--	100	--	100	--
1984	250	--	250	--	250	--	250	--
1985	650	24	610	27	620	36	640	28
1986	700	49	680	62	680	76	700	66
1987	770	49	770	62	760	76	780	66
1988	780	49	790	62	770	76	800	66
1989	780	49	800	61	770	76	800	66
1990	800	210	830	250	820	300	820	370
1991	800	210	840	260	830	310	860	390
1992	870	210	850	260	840	310	870	390
1993	870	210	910	260	900	310	930	390
1994	920	210	990	260	960	310	1000	390
1995	930	220	1000	280	960	320	1000	410
1996	930	220	1000	290	960	330	1000	420
1997	930	220	1000	290	960	330	1000	420
1998	990	220	1100	290	--	--	1100	420
1999	1000	220	1200	290	--	--	1200	420
2000	--	--	1300	320	--	--	1200	460
2001	--	--	1300	330	--	--	1200	470
2002	--	--	1300	330	--	--	1200	470
2003	--	--	1300	330	--	--	--	--
2004	--	--	1400	330	--	--	--	--
2005	--	--	--	--	--	--	--	--
	Man Yrs/Yr Fuel Cycle IIb & III							
1980	60	--	60	--	60	--	60	--
1981	65	--	65	--	65	--	65	--
1982	80	--	80	--	80	--	80	--
1983	100	--	100	--	100	--	100	--
1984	250	--	250	--	250	--	250	--
1985	690	24	710	27	670	36	730	28
1986	770	49	780	61	740	76	830	67
1987	850	49	860	61	810	76	930	67
1988	870	49	880	61	830	76	960	65
1989	870	49	890	61	840	76	960	65
1990	900	180	920	250	870	290	1000	370
1991	910	180	930	260	900	310	1000	380
1992	920	180	940	260	920	310	1000	380
1993	980	180	1000	260	990	310	1100	380
1994	1100	180	1100	260	1100	310	1200	380
1995	1100	190	1100	280	1100	320	1200	410
1996	1100	190	1100	300	1100	340	1200	430
1997	1100	190	1100	300	1100	340	1200	430
1998	1200	190	1200	300	--	--	1300	430
1999	1200	190	1300	300	--	--	1300	430
2000	1300	200	1300	320	--	--	1400	460
2001	1300	210	1300	330	--	--	1400	470
2002	1300	210	1300	330	--	--	1400	470
2003	1400	210	1400	330	--	--	--	--
2004[c]	1400	210	1500	330	--	--	--	--
2005[c]	1500	220	--	--	--	--	--	--

a. Operations includes surface and subsurface waste handling and emplacement, general operation support (maintenance, utilities, etc.) and administrative personnel.
b. Drilling and backfill includes hole drilling, trench excavation, sleeve emplacement, and backfill.
c. Years 2004 and 2005 apply to fuel cycle IIb only.

Supplies and Utilities

Repository supply and utility requirements are listed in Tables 7.5.9 and 7.5.10, respectively.

TABLE 7.5.9. Supply Requirements

Description	Use	Average Annual Requirement Salt	Granite	Shale	Basalt
Fuel Cycle IIa					
HLW canister overpacks[a,b]	Contains damaged HLW canisters	1	3	3	4
FRW/ILW canister overpacks[a]	Contains damaged FRW/ILW canisters	2	2	2	2
ILW drum-packs	Contains 3 ILW drums	6000	7100	5700	6700
HLW retrievability sleeves[b,c]	Emplacement hole liner	270	470	610	610
FRW/ILW retrievability sleeves	Emplacement hole liner	5200[c]	8600[d]	5200[c]	8100[d]
HLW concrete plug[c]	Emplacement hole plug	270	470	610	610
FRW/ILW concrete plug[c]	Emplacement hole plug	5200	5200	5200	5200
FRW/ILW trench rack[d]	Sleeve support during trenching	--	2850[e]	--	2700[e]
Fuel Cycle IIb					
HLW canister overpacks[a,b]	Contains damaged HLW canisters	2	3	3	4
FRW/ILW canister overpacks[a]	Contains damaged FRW/ILW canisters	2	2	2	2
ILW drum-packs	Contains 3 ILW drums	7500	7200	5700	6800
HLW retrievability sleeves[b,c]	Emplacement hole liner	270	390	610	550
FRW/ILW retrievability sleeves	Emplacement hole liner	5300[c]	8600[d]	5300[c]	8200[d]
HLW concrete plug[c]	Emplacement hole plug	270	390	610	550
FRW/ILW concrete plug[c]	Emplacement hole plug	5300	5300	5300	5300
FRW/ILW trench rack[d]	Sleeve support during trenching	--	2880[e]	--	2720[e]
Fuel Cycle III					
HLW canister overpacks[a,b]	Contains damaged HLW canisters	2	3	3	4
FRW/ILW canister overpacks[a]	Contains damaged FRW/ILW canisters	2	2	2	2
ILW drum-packs	Contains 3 ILW drums	7000	7200	5700	6800
HLW retrievability sleeves[b,c]	Emplacement hole liner	270	390	610	550
FRW/ILW retrievability sleeves	Emplacement hole liner	5300[c]	8600[d]	5300[c]	8200[d]
HLW concrete plug[c]	Emplacement hole plug	270	390	610	550
FRW/ILW concrete plug[c]	Emplacement hole plug	5300	5300	5300	5300
FRW/ILW trench rack[d]	Sleeve support during trenching	--	2880[e]	--	2720[e]

a. Overpack requirements are based on 0.1% of canisters received leaking or damaged.
b. HLW canister diameter changes with time as neccessary to maintain canister heat output within limits (Section 7.3). The resulting overpack and sleeve sizes are:

Canister Size	Over-Pack Size	Sleeve Size
31 cm dia x 3 m long	36 cm dia x 3.2 m long	41 cm dia x 6.1 m long
25 cm dia x 3 m long	31 cm dia x 3.2 m long	36 cm dia x 6.1 m long
20 cm dia x 3 m long	25 cm dia x 3.2 m long	31 cm dia x 6.1 m long
15 cm dia x 3 m long	20 cm dia x 3.2 m long	25 cm dia x 6.1 m long

c. Sleeves and plugs for ready retrievabilty are needed for first 5 years only.
d. FRW and ILW are emplaced in trenches that require continual use of sleeves and racks. Therefore annual quantities of FRW/ILW sleeves and racks in granite and basalt are for all years of operation.
e. Each rack holds 3 FRW/ILW canisters or overpacks.

TABLE 7.5.10. Utility Requirements[a]

Utility	Use Rate	Annual Requirement
Coal	8.2 MT/hr	>2,000 MT/yr
Electricity		
Salt	13 MW	110,000 MWh
Granite	15 MW	130,000 MWh
Shale	13 MW	110,000 MWh
Basalt	15 MW	130,000 MWh
Diesel fuel	1.5 m^3/hr	13,000 m^3/yr
Steam	91 MT/hr	800,000 MT/yr
Water	2 kg/hr	10,000 kg/yr

a. Peak yearly requirements.

7.5.6 Secondary Wastes

Secondary wastes generated at the repositories during the handling and emplacement of wastes are treated and disposed of onsite as described in Section 7.4.6 for the once-through fuel cycle repository.

7.5.7 Emissions

All wastes arriving at the repository are fully contained in leak tested steel canisters or steel drums. As a result, the only sources for airborne emissions from these waste containers are handling accidents that could damage and breach the canisters. Potential accidents are described in Section 7.5.9.

An estimate of the integrated annual release due to minor accidents (Section 7.4.9) for this facility is included in Table 7.5.11 in addition to annual emissions from onsite combustion of coal and diesel and the ventilation system cooling tower. The integrated annual release was developed by weighing the minor accident releases by their expected frequencies and summing the quantities for all identified minor accidents. In addition, a contingency was included in the integrated release to account for unidentified minor accidents and to compensate for the uncertainty in expected frequency information.

7.5.8 Decommissioning Considerations

Decommissioning considerations for a geologic repository are discussed in Section 7.4.8.

7.5.9 Postulated Accidents

All structures are maintained at a negative pressure relative to the atmosphere, and all entries into and exits from confinement areas are made through air locks. Contamination is controlled by directing air flow from areas of less contamination potential to areas of increasing

TABLE 7.5.11. Emissions

Emission	Description	Annual Quantity	Radioactivity Release Factor[a]
Gaseous	Coal and Diesel Combustion	SO_x - 610 MT CO_x - 150 MT Hydrocarbons - 54 MT NO_x - 940 MT Particulates - 27 MT Heat - 2.4×10^7 MJ	
	Minor Accident Integrated Annual Release		3.6×10^{-12}
Cooling Tower Water	Evaporated (T = 38°C)	9.2×10^7 kg	
	Drift (T = 38°C)	4.4×10^5 kg	
	Blowdown (T = 27°C)	1.6×10^7 kg	

a. Fraction of activity (Table 5.3.1) in one drum of LLW released to atmosphere during year of maximum LLW receiving rate. Includes air filtration system DF(10^7).

contamination potential. Air discharged from confinement areas is exhausted through a prefilter and two high-efficiency particulate air (HEPA) filters. Ventilation systems are backed up by standby facilities to maintain confinement in the event of fan breakdown, filter failure, or normal power outage. Automatic monitoring of all potential sources of contaminated effluents is provided with remote readout and alarm at both the central control room in the mine operations building and the guardhouse.

Repository site selection considerations, discussed in Section 7.2.1, are general geologic criteria for acceptable repository locations. These factors provide maximum assurance that the repository integrity will be maintained throughout the hazardous lifetime of the waste.

Postulated minor and moderate accident scenarios for the repositories are given in Tables 7.5.12 and 7.5.13. No accidents that could be classified as severe accidents could be realistically postulated for this technology. Several extremely improbable nondesign basis accidents are described in Table 7.5.14.

The accident descriptions are similar for all geologic media considered except for 7.7 (tornado strikes salt pile) and 7.13 (repository breach by solution mining) that are posulated only in salt. However, thc amount of wastes handled and emplaced does vary with the selected medium. These variations are considered in the release column for the source terms.

Minor and moderate accidents considered in this section are related to the operational phase of the repository. They postulate various mechanical and thermal environments to which the wastes may be subjected. The release fractions for these accidents were developed using energy absorption models and conservative assumptions on waste dispersion.

Secondary wastes generated onsite are incinerated and immobilized before emplacement. Accidents for these processes are presented in Sections 4.3, 4.4. and 4.7 of this report.

TABLE 7.5.12. Postulated Minor Accidents - Reprocessing Fuel Cycles

Accident No. and Description	Sequence of Events	Safety System	Release
7.1 - LLW drum rupture due to handling error. Estimated frequency $\sim 3 \times 10^{-5}$ per received drum.	1. Operator error may cause the fork from fork-lift truck to penetrate a 55 gal drum. 2. Drum contents are exposed to the facility atmosphere. 3. Drum contents repackaged. 4. Area decontaminated.	1. Waste in massive form which would resist puncture. 2. Repackaging and decontamination facilities available.	A release fraction of 2.5×10^{-5} to the mine atmosphere from one drum is estimated.
7.2 - Minor canister failure.	1. Rough handling during transportation and unloading or canister defect results in the formation of a pin hole leak. 2. Leak detected in receiving facility. 3. Canister overpacked and placed in storage.	1. Canisters inspected prior to shipment. 2. Canisters pressurized with helium for leak detection. 3. Overpacking facilities available.	High-level Waste - Pin holes are postulated to occur in 0.1% of the waste canisters. However, a mechanism of suspending the waste in the void volume was not identified. Therefore it was assumed that releases from this type failure is essentially zero. FRW, ILW and LLW - Canisters and drums filled with fixed waste have no internal pressure to drive a release. They are also in a massive form which would reduce fines formation in the void volume. For these reasons it is assumed that releases from this type failure are essentially zero.
7.3 - Receipt of externally contaminated canister.	1. Canister received with smearable contamination above specification limits. 2. Contamination detected. 3. Canister decontaminated and placed in storage.	1. Canisters inspected prior to shipment. 2. Canister inspected on receipt. 3. Decontamination facilities available.	None
7.4 - Dropped shipping cask.	1. Equipment failure or operator error drops shipping cask into transfer gallery. 2. Cask inspected. Expected to be undamaged. 3. Impact absorber removed and replaced.	1. Impact absorber minimizes cask damage.	None

TABLE 7.5.13. Postulated Moderate Accidents - Reprocessing Fuel Cycles

Accident No. and Description	Sequence of Events	Safety System	Release
7.5 - Waste container drop during handling.	1. Equipment failure or operator error results drop sufficient to fail waste container. 2. Particulate release to cell filters. 3. Container contents repackaged. 4. Area decontaminated.	1. Drop height minimized by facility design. 2. Cell HEPA filters reduce release to atmosphere. 3. Repackaging and decontamination facilities available.	The maximum drop height for any waste form is 20 m. This is much less than a drop down the mine shaft. Therefore it is expected that releases from this accident would be much less than Accident 7.6.
7.6 - Waste package dropped down mine shaft. Estimated frequency: HLW 7×10^{-7}/yr FRW 2×10^{-6}/yr ILW 2×10^{-5}/yr LLW 3×10^{-6}/yr	1. Canistered waste shaft hoist fails. 2. Hoist cage containing canisters to mine level. 3. Canister breach on impact. 4. Canister contents repackaged. 5. Area decontaminated.	1. Failsafe wedge type braking system on hoist cage. 2. Mine exhaust filter system reduces atmospheric releases. 3. Repackaging and decontamination equipment available.	HLW - Four canisters are lowered in cage. Release fraction of 6×10^{-3} (<10 μm particles) to mine atmosphere. Use 3 MTHM equivalent per canister and waste radionuclide tables: Cycle IIa: 3.3.8, 3.3.13 Cycle IIb: 3.3.8, 3.3.12 Cycle III: 3.3.9, 3.3.14. FRW - One canister is lowered in cage. Release fraction of 1×10^{-3} (<10 μm particles, Table 3.3.7) to mine atmosphere. ILW - One canister containing three drums is lowered in cage. Release fraction of 0.1 (<10 μm particles, Table 5.3.1) to mine atmosphere. LLW - Twelve drums lowered in cage. Release fraction of 0.2 (<10 μm, Table 5.3.1) to mine atmosphere.

TABLE 7.5.13. (contd)

Accident No. and Description	Sequence of Events	Safety System	Release
7.7 – Tornado strikes mined salt storage area. Estimated frequency $8 \times 10^{-5}/km^2$-yr.	1. Tornado strikes mined salt storage area. 2. Salt dispersed to atmosphere.	1. Site selection criteria limit maximum credible tornado. 2. Salt is covered as it accumulates.	Tornado strikes when pile contains maximum amount of salt. Pile is 1 km wide at bottom, 910 m wide at top, 30 m tall and 940 m long. 3.6×10^7 MT is total salt stored in pile. Only 7×10^4 m^2 is uncovered and available for dispersion. 2.2×10^4 MT (1%) is removed by tornado.
7.8 - LLW drum rupture due to mechanical damage and fire.	1. Fuel tank on fork-lift or other service explodes. 2. Drums of LLW rupture and are exposed to fire. 3. Particulate release to mine filters. 4. Waste repackaged. 5. Area decontaminated.	1. Fuel source limited by tank size. 2. Wastes in massive form which resists dispersion. 3. Release to atmosphere reduced by HEPA filters. 4. Fire fighting equipment available in mine. 5. Repackaging and decontamination equipment available.	It is estimated that no more than 10% of the activity (see Table 5.3.1) in the twelve drums would reach the cell filters because of this accident. Therefore, the releases from this accident are less than the releases from Accident 7.6.
7.9 - LLW drum rupture due to internal explosion.	1. Small amounts of highly reactive or explosive material in waste. 2. Vigorous reaction results in explosion of a waste drum. 3. Particulate release to cell filters. 4. Drum contents repackaged. 5. Area decontaminated.	1. Explosive chemicals are not used in waste generation processes. 2. Waste form designed to prevent formation of explosive material or vigorous reactions. 3. Release to stack reduced by HEPA filters. 4. Repackaging and decontamination equipment available.	No more than one half (see Table 5.3.1) of the contents in one LLW drum would be released to the mine filters. The releases from this accident are less than the releases from Accident 7.6.

TABLE 7.5.14. Postulated Non-Design Basis Accidents - Reprocessing Fuel Cycle

Accident No. and Description	Sequence of Events	Safety System	Release
7.10 - Nuclear warfare.	1. 50-megaton nuclear weapon bursts on surface above repository. 2. Crater formed to 340 m, with fracture zone to 500 m.	1. Repository depth of 540 m - Salt 590 m - Granite 420 m - Shale 560 m - Basalt	Although the fracture zone reaches the repository in shale, releases are expected to be less than those from Accident 7.11 (repository breach by meteorite).
7.11 - Repository breach by meteorite. Expected frequency $\sim 2 \times 10^{-13}$ per year.	1. Meteor with sufficient mass and velocity to form 2 km dia crater impacts repository area after closure. 2. Crater extends to waste horizon, dispersing waste to atmosphere. 3. Crater partially refilled with rubble covering repository.	1. Repository depth.	One percent (1%) of the inventory (Table 7.5.4) is released on impact with one half going to local fallout and one half going to stratospheric dispersion. Waste description tables: Cycle IIa: 3.3.8, 3.3.13 Cycle IIb: 3.3.8, 3.3.12 Cycle III: 3.3.9, 3.3.14.
7.12 - Repository breach by drilling.	1. Societal changes lead to loss of repository records and location markers. 2. Drilling occurs.	1. Repository depth. 2. Repository location monuments and records. 3. Site criteria--no desirable resources.	Drilling may occur anywhere in repository. Probability of contacting a contaminated zone and/or canister (within 30 cm radius of canister), 0.005. One-fourth of a canister is brought to the surface in the drilling mud. The activity is uniformly distributed over 0.5 ha (1.2 acres) in the top 5 cm (2 inches) of soil. Waste description tables: Cycle IIa: 3.3.8, 3.3.13 Cycle IIb: 3.3.8, 3.3.12 Cycle III: 3.3.9, 3.3.14.
7.13 - Repository breach by solution mining (salt only).	1. Societal changes lead to loss of records and location markers. 2. Exploratory drilling (see Accident 7.12) leads to the discovery of salt.	1. Monuments mark repository location. 2. Repository depth of 540 m. 3. Site criteria exclude areas with desirable resources.	As salt is dissolved and carried away, water comes into contact with exposed wastes. At a leach rate of 1×10^{-5} g/cm^2 day, 4.7 MTHM equivalent waste for fuel cycles IIb and III and 2.8 MTHM for fuel cycle IIa are

TABLE 7.5.14. contd

Accident No. and Description	Sequence of Events	Safety System	Release
	3. Salt is mined using solution extraction techniques. 4. Contamination is discovered after 1 year and mining is discontinued.	4. Other plentiful and accessible salt deposits are available.	leached into the brine during the first year of solution mining. Waste description tables: Cycle IIa: 3.3.8, 3.3.13 Cycle IIb: 3.3.8, 3.3.12 Cycle III: 3.3.9, 3.3.14.
7.14 - Volcanism	1. Volcanic activity at repository carries wastes to surface.	1. Site criteria - no history or potential for volcanic activity.	Release equal to or less than Accident 7.15.
7.15 - Faulting and groundwater transport. Estimated frequency 2×10^{-13} per year.	1. Fault intersects repository. 2. Access is created between high-pressure aquifer, waste and surface. 3. Aquifer carries wastes to surface.	1. Site criteria--low seismic risk zone. 2. Site criteria--minimal groundwater. 3. Repository depth. 4. Low leachable waste forms.	The following fractions of the repositories' contents (Table 7.5.4) are available for leaching. Cycle IIa Salt - 7×10^{-3} Granite - 7×10^{-3} Shale - 3×10^{-3} Basalt - 9×10^{-3} Tables 3.3.7, 3.3.8, 3.3.13, and 5.3.1. Cycle IIb Salt - 7×10^{-3} Granite - 7×10^{-3} Shale - 7×10^{-3} Basalt - 1×10^{-2} Tables 3.3.7, 3.3.8, 3.3.12, and 5.3.1. Cycle III Salt - 7×10^{-3} Granite - 7×10^{-3} Shale - 7×10^{-3} Basalt - 1×10^{-2} Tables 3.3.7, 3.3.9, 3.3.14, and 5.3.1.
7.16 - Erosion	1. Repository overburden subject to high erosion.	1. Site criteria--low erosion rates. 2. Repository depth.	Release equal to or less than Accident 7.11.
7.17 - Criticality	1. Fault or groundwater action leaches Plutonium from Cycle IIa HLW. 2. Leached Pu accumulates into critical mass.	1. Site criteria--low seismic risk zone. 2. Site criteria--minimal groundwater. 3. Repository depth.	None

All wastes lowered into the repositories are assumed to be noncombustible. If combustibles were allowed, it is expected that no great increase in accidental releases would occur. A fire in drummed wastes is rare and generates relatively small amounts of heat. This is expected to limit the fire to a single drum. Releases from this accident would be less than that of 7.9 (LLW drum rupture due to internal explosion).

Accidents for the isolation phase of the repository operation (non-design basis) are unsophisticated in their description of geologic processes, but are believed to provide conservative estimates of radionuclide releases.

For purposes of environmental consequence analysis, the material release associated with accidents numbered 7.2, 7.6, 7.7, 7.11, 7.12, 7.13, and 7.15 in Tables 7.5.12, 7.5.13, and 7.5.14 have been selected as umbrella source terms (the concept of an umbrella source term is explained in Section 3.7 - Basis for Accident Analysis). This means that the releases from these accidents are the largest in their respective source term categories. The environmental consequences of these accidents are described in DOE/ET-0029.[3]

7.5.10 Costs - Recycle Repositories

The following section describes the methods used for developing cost estimates for repositories in geologic media for reprocessing fuel cycles. Except when noted, the methodology is the same as that described in Section 7.4.10 for the once-through fuel cycle. All estimates are in constant mid-1976 dollars.

7.5.10.1 Construction Costs for Fuel Recycle Repositories

Construction costs were estimated in the same manner described in Section 7.4.10. Detailed construction cost estimates for three LWR reprocessing fuel cycles are given in Appendix 7B in Tables 7.B.1 through 7.B.24. The three reprocessing fuel cycles are uranium-only recycle with plutonium disposal in HLW (IIa), uranium-only recycle with plutonium storage (IIb) and uranium-plutonium recycle (III). Table 7.5.15 summarizes the total construction and mining costs for the reference accelerated mining alternative for repositories in salt, granite, shale and basalt. Costs for repositories in granite and basalt are about 70 to 100% more than those in salt and shale because of higher mining costs in these media. Cost differences between fuel cycles are relatively insignificant. Table 7.5.16 gives the construction cost summary for the continuous mining alternative. The construction costs for continuous mining are only slightly lower than those for accelerated mining (reasons for this are discussed in Section 7.4.10.1).

The construction and mining cash flow schedules were derived by applying the expenditure schedules in Tables 7.4.15 through 7.4.16 for surface facilities; shafts, hoists and underground facilities; and mining respectively to the cost estimates in Tables 7.B.1 through 7.B.24. The backfilling expenditure schedule is given in Table 7.5.17. For the continuous mining alternative, the mining expenditure schedules for the reprocessing fuel cycles are shown in Appendix 7B in Tables 7.B.25 through 7.B.30.

TABLE 7.5.15. Total Repository Construction Costs by Expenditure Type and Geologic Media for Reprocessing Fuel Cycles - Accelerated Mining, Millions of 1976 Dollars

Fuel Cycle and Geologic Media	Construction	Mining	Backfilling	Total
IIa				
Salt	480	380	70	930
Granite	540	1,060	120	1,720
Shale	450	550	70	1,070
Basalt	570	1,250	140	1,960
IIb				
Salt	490	430	80	1,000
Granite	550	1,060	120	1,730
Shale	460	550	70	1,080
Basalt	580	1,210	140	1,930
III				
Salt	490	440	80	1,010
Granite	560	1,070	120	1,750
Shale	460	550	70	1,080
Basalt	580	1,250	140	1,970

TABLE 7.5.16. Total Repository Construction Costs by Expenditure Type and Geologic Media for Reprocessing Fuel Cycles - Continuous Mining, Millions of 1976 Dollars

Fuel Cycle and Geologic Media	Construction	Mining	Backfilling	Total
IIa				
Salt	450	380	50	880
Granite	490	1,070	100	1,660
Shale	420	550	60	1,030
Basalt	530	1,250	110	1,890
IIb				
Salt	460	430	40	930
Granite	500	1,070	100	1,670
Shale	430	550	60	1,040
Basalt	540	1,210	110	1,860
III				
Salt	460	440	40	940
Granite	500	1,080	100	1,680
Shale	430	550	60	1,040
Basalt	550	1,250	110	1,910

TABLE 7.5.17. Repository Expenditure Schedule for Backfilling for Recycle Repositories

Media	Fuel Cycle	Years	Materials and Labor Percent per Year
Salt	U-only recycle - Pu in waste	10	10.0
	U-only recycle - Pu stored	16	6.25
	U-Pu recycle	14	7.15
Granite	All recycle cases	15	6.67
Shale	All recycle cases	8	12.5
Basalt	All recycle cases	13	7.7

The construction cash flows were allocated by waste type using the data for the reprocessing fuel cycles in Tables 7.C.1 through 7.C.8 in Appendix 7C. Canistered waste costs were allocated to each canistered waste type (HLW, FRW and ILW) by the ratio of the number of canisters of each waste type received to the total number of canisters received. Nonallocable costs including nonallocated shaft costs were allocated to each waste type by the ratio of containers received to total containers received.

7.5.10.2 Operating Costs for Fuel Recycle Repositories

Operating costs were developed for reprocessing waste repositories in the same manner as described in Section 7.4.10. Waste receipt rates, manpower, supplies, and utilities requirements are given in Tables 7.5.2 and 7.5.8 through 7.5.10 respectively. The jobcategory percentages, unit costs of overpacks, sleeves and racks, hole drilling, and trenching costs are given in Tables 7.5.18 through 7.5.20. Unit costs of sleeve emplacement are the same as for the once-through cycle. An additional 25% contingency allowance for miscellaneous and unidentified costs was added to the operating cost estimates. The operating costs shown in Table 7.5.21 are total operating costs over the entire operating period of the repository. Labor comprises the largest portion of the total cost for salt and shale repositories. Materials costs are substantial in granite and basalt repositories due to sleeve and rack requirements for FRW/ILW canisters. The most significant causes of the variations in operating costs between fuel cycles and repository media are the differences in repository capacities and the trenching method of FRW/ILW waste emplacement in granite and basalt. A comparison of operating costs on a unit basis is made in the levelized unit cost section.

7.5.10.3 Backfilling Costs for Fuel Recycle Repositories

Backfilling costs are listed in Table 7.5.15. The expenditure schedule for backfilling is given in Table 7.5.17 and assumes backfilling starts in the sixth year following start of waste emplacement operations.

7.5.10.4 Decommissioning and Shaft Sealing Costs for Fuel Recycle Repositories

Decommissioning and shaft sealing costs are estimated in the same manner as explained in Section 7.4.10. The estimated costs are listed in Table 7.5.22.

TABLE 7.5.18. Percentage of Total Repository Operating Personnel by Job Category and Repository Type - All Reprocessing Fuel Cycles

Job Category	Salt	Granite	Shale	Basalt
Underground Storage				
Canistered waste	9.2	11.7	10.5	12.0
Low-level waste	1.8	1.9	1.7	1.8
Above Ground Storage				
Canistered waste building	21.3	26.5	24.0	27.5
- Low-level waste building	2.5	2.7	2.3	2.8
Control Room	3.1	3.6	3.5	3.7
Railroad Operations	5.7	5.6	5.9	5.6
Shipment Inspections	2.0	1.9	1.8	1.9
Liquid Rad Waste	1.7	2.2	1.9	2.2
Ventilation Maintenance	1.3	1.0	1.2	1.0
Utilities Support	2.2	1.8	2.1	1.7
Motor Pool	0.8	0.7	0.8	0.6
Maintenance	6.2	4.9	5.7	4.7
Security	14.4	11.6	13.3	10.9
Management	17.7	15.9	16.1	15.8
Other	10.1	8.0	9.2	7.8
	100.0	100.0	100.0	100.0

TABLE 7.5.19. Unit Costs for Overpack, Racks, Sleeves and Plugs - Recycle Repositories

Description	Unit Cost, 1976 Dollars
HLW Overpack	400
ILW/FRW Canister or Drum (3) Overpack	1,050
LLW Drum Overpack	200
HLW Retrievability Sleeve and Plug	
for 30 cm diameter HLW Canister	700
for 25 cm diameter HLW Canister	610
for 20 cm diameter HLW Canister	520
for 15 cm diameter HLW Canister	430
ILW/FRW Retrievability Sleeve and Plug	1,550
Trench Rack for 3 ILW/FRW Canisters	2,500

TABLE 7.5.20. Unit Costs and Manpower Requirements for Hole Drilling and Trenching - Recycle Repositories

Media	Category	Retrievable Emplacement	Unit Cost, (1976 Dollars) HLW with Canister Diameter of: (15 cm)	(20 cm)	(25 cm)	(30 cm)	FRW	ILW[a]
Salt	Hole Drilling	No	230	240	250	280	460	460
		Yes	240	250	280	300	500	500
Granite	Hole Drilling	No	1,500	1,800	2,100	2,400	--	--
		Yes	1,800	2,100	2,400	3,000	--	--
	Trenching	--	--	--	--	--	570	370
Shale	Hole Drilling	No	480	510	550	580	980	980
		Yes	510	550	580	640	1,040	1,040
Basalt	Hole Drilling	No	1,580	1,890	2,200	2,530	--	--
		Yes	1,890	2,200	2,530	3,150	--	--
	Trenching	--	--	--	--	--	590	370
			Man-hours per Canister					
Salt	Hole Drilling	No	4	4	5	5	8	8
		Yes	4	5	5	6	8	8
Granite	Hole Drilling	No	19	23	27	31	--	--
		Yes	23	27	31	39	--	--
	Trenching	--	--	--	--	--	10	10
Shale	Hole Drilling	No	8	9	10	10	16	16
		Yes	9	10	10	11	16	16
Basalt	Hole Drilling	No	21	26	30	34	--	--
		Yes	26	30	34	43	--	--
	Trenching	--	--	--	--	--	11	11

a. Assuming 3 55-gal ILW drums per canister.

TABLE 7.5.21. Total Operating Costs for Repositories Storing Reprocessing Wastes, Millions of 1976 Dollars

Fuel Cycle	Operating Expenditure	Repository Type: Salt	Granite	Shale	Basalt
Uranium-only Recycle, Pu in HLW	Labor	260	410	220	350
	Materials	140	570	120	480
	Utilities	60	90	50	80
	Hole drilling/trenching	60	140	110	130
	Sleeve placement	5	20	5	20
	Overhead	40	60	40	50
	Contingencies	145	320	135	280
	Total	710	1610	680	1390
Uranium-only Recycle, Pu stored	Labor	410	410	220	350
	Materials	215	510	120	490
	Utilities	80	90	50	80
	Hole drilling/trenching	100	140	110	120
	Sleeve placement	5	20	5	20
	Overhead	60	60	40	50
	Contingencies	220	320	135	280
	Total	1090	1610	680	1390
Uranium-Plutonium Recycle	Labor	420	460	240	420
	Materials	190	570	130	490
	Utilities	70	80	50	80
	Hole drilling/trenching	80	140	110	130
	Sleeve placement	5	20	5	20
	Overhead	60	60	35	50
	Contingencies	205	330	140	300
	Total	1030	1660	710	1490

TABLE 7.5.22. Decommissioning and Shaft Sealing Costs for Recycle Repositories

Fuel Cycle	$1,000,000s (mid-1976) Salt	Granite	Shale	Basalt
IIa	24.5	23.9	24.3	23.8
IIb	25.2	24.6	25.0	24.7
III	25.2	24.6	25.0	24.6

7.5.10.5 Levelized Unit Costs for Recycle Repositories

Levelized unit costs for each waste type in the reprocessing fuel cycles (accelerated mining) are given in Tables 7.5.23 through 7.5.25. These costs are calculated from the construction, mining, and operating cash flows previously developed and the annual waste receipt rate in kilograms of equivalent heavy metal. Levelized costs for the continuous mining case were not developed; the effect on levelized costs would be similar to the effect in the once-through cycle described in Section 7.4.10.5.

TABLE 7.5.23. Levelized Unit Cost Estimate for Uranium-Only Recycle Repository - Pu in HLW - Accelerated Mining, Dollars per kg HM (1976)

Geologic Media	Waste Type	Levelized Construction and Mining Cost	Levelized Operating Cost	Total Levelized Unit Cost[(a)]	Levelized Costs for 0 to 10% Cost of Money
Salt	HLW	21.90	5.50	27.40	17 - 34
	FRW	1.30	1.00	2.30	1.70 - 2.80
	ILW	9.60	6.60	16.20	12 - 19
	LLW	1.80	0.40	2.20	1.40 - 2.60
	Total	34.60	13.50	48.10	32 - 58
Granite	HLW	27.90	4.90	32.80	17 - 43
	FRW	1.90	2.10	4.00	2.90 - 4.60
	ILW	11.90	12.70	24.60	17 - 28
	LLW	2.80	0.40	3.20	1.80 - 4.10
	Total	44.50	20.10	64.60	39 - 80
Shale	HLW	29.70	5.40	35.10	23 - 43
	FRW	1.80	1.20	3.00	2.30 - 3.60
	ILW	12.10	8.20	20.30	15 - 24
	LLW	2.60	0.40	3.00	2.00 - 3.50
	Total	46.20	15.20	61.40	42 - 74
Basalt	HLW	37.60	5.80	43.40	24 - 56
	FRW	2.10	2.10	4.20	3.10 - 4.90
	ILW	14.10	12.60	26.70	19 - 30
	LLW	3.60	0.40	4.00	2.40 - 4.90
	Total	57.40	20.90	78.30	49 - 96

a. Assuming a cost of money of 7%.

TABLE 7.5.24. Levelized Unit Cost Estimate for Uranium-Only Recycle Repository - Pu Stored - Accelerated Mining, Dollars per kg HM

Geologic Media	Waste Type	Levelized Construction and Mining Cost	Levelized Operating Cost	Total Levelized Unit Cost	Levelized Costs for 0 to 10% Cost of Money
Salt	HLW	13.70	3.40	17.10	9 - 23
	FRW	1.30	1.00	2.30	1.50 - 2.90
	ILW	8.10	6.60	14.70	10 - 18
	LLW	1.40	0.40	1.80	1.10 - 2.20
	Total	24.50	11.40	35.90	22 - 46
Granite	HLW	27.80	4.80	32.60	17 - 43
	FRW	1.90	2.10	4.00	2.90 - 4.60
	ILW	12.20	12.70	24.90	17 - 28
	LLW	2.80	0.40	3.20	1.80 - 4.10
	Total	44.70	20.00	64.70	39 - 80
Shale	HLW	29.60	5.30	34.90	23 - 43
	FRW	1.80	1.20	3.00	2.30 - 3.60
	ILW	12.50	8.30	20.80	16 - 24
	LLW	2.60	0.40	3.00	2.00 - 3.50
	Total	46.50	15.20	61.70	43 - 74
Basalt	HLW	36.30	5.40	41.70	23 - 54
	FRW	2.30	2.10	4.40	3.20 - 5.10
	ILW	14.30	12.70	27.00	19 - 31
	LLW	3.50	0.40	3.90	2.30 - 4.90
	Total	56.40	20.60	77.00	48 - 95

TABLE 7.5.25. Levelized Unit Cost Estimate for Uranium - Plutonium Recycle Repository - Accelerated Mining, Dollars per kg HM

Geologic Media	Waste Type	Levelized Construction and Mining Cost	Levelized Operating Cost	Total Levelized Unit Cost	Levelized Costs for 0 to 10% Cost of Money
Salt	HLW	16.20	4.20	20.40	12 - 27
	FRW	1.30	1.10	2.40	1.70 - 3.00
	ILW	8.80	6.90	15.70	10 - 18
	LLW	1.60	0.50	2.10	1.40 - 2.60
	Total	27.90	12.70	40.60	25 - 51
Granite	HLW	28.00	4.90	32.90	18 - 43
	FRW	1.90	2.20	4.10	3.00 - 4.70
	ILW	12.20	13.10	25.30	18 - 29
	LLW	3.00	0.60	3.60	2.10 - 4.50
	Total	45.10	20.80	65.90	41 - 81
Shale	HLW	29.40	5.60	35.00	22 - 42
	FRW	1.80	1.30	3.10	2.40 - 3.70
	ILW	12.30	8.50	20.80	16 - 24
	LLW	2.70	0.60	3.30	2.30 - 3.90
	Total	46.20	16.00	62.20	43 - 74
Basalt	HLW	37.10	6.00	43.10	24 - 55
	FRW	2.10	2.30	4.40	3.20 - 5.00
	ILW	14.40	13.30	27.70	20 - 32
	LLW	3.70	0.60	4.30	2.70 - 5.40
	Total	57.30	22.20	79.50	50 - 97

A comparison of costs reveals the following:

- Salt is the least expensive disposal medium when comparing total unit costs. Granite costs about 20-80% more, shale 30-90% more, basalt 50-120% more than disposal in salt.
- The differences are small between the three fuel cycles when comparing total unit costs for each waste type in each disposal medium except in the case of HLW disposal in salt repositories. When plutonium is discarded in the HLW, the HLW disposal costs in salt repositories are increased because thermal criteria limit the waste emplacement density (Table 7.3.4).
- HLW accounts for about 50-55% of total disposal costs, ILW accounts for 35-40%, and fuels residue (cladding and fuel assembly hardware) and LLW account for about 5% each. account for about 5% each.
- Construction, mining and interest charges account for 70-75% of the total cost and they also account for most of the differences in disposal costs for different media. Unit operating costs in salt and shale are somewhat lower than in the other media.
- Costs are strongly affected by variations in the cost of money as shown in the last columns of Tables 7.5.23 through 7.5.25.
- Because of the uncertainty in the estimating procedures, the uncertainty in the possible mining procedures, and the uncertainty in the cost of money, the overall uncertainty in the total unit cost estimates is estimated to be ±50%.

The unit costs in Tables 7.5.23 through 7.5.25 are translated in Table 7.5.26, to costs per container for the reference containers used in this study.

7.5.11 Construction Activities and Requirements

Many of the activities relating to site preparation and facility construction may have some impact on the environment, the local economy, and the natural resources surrounding the area. The information which follows provides a basis for evaluating the impact of construction activities. These activities include construction of surface facilities, shafts, and all mining activities.

7.5.11.1 Construction Activities

The engineering and construction schedule for this project is dependent on the sequence of site evaluation and acquisition, federal approvals, funding and other uncertainties. The schedule is further complicated by technical considerations including the coordination of underground and surface construction activities. For this report it is assumed that an environmental impact statement and preliminary safety analysis report are prepared and are subjected to an approval procedure similar to that required for a commercial facility. It is also assumed that a suitable site has been pre-selected and subjected to geologic and safety evaluations before beginning construction operations, and that no unanticipated underground conditions are encountered in the course of shaft sinking and mining. Figure 7.5.13 summarizes the overall project schedule for the engineering, procurement, and construction phases of the project.

TABLE 7.5.26. Levelized Unit Costs for Reprocessing Fuel Cycles in Terms of Dollars per Container

Geologic Media	Waste Type	Container Type[a]	$ per Container (mid-1976 $) Uranium-Only Recycles Pu in HLW	Pu Stored	Uranium-Plutonium Recycle
Salt	HLW[b]	All HLW canisters	64,200	52,100	47,700
	FRW	76 cm canister	9,600	9,600	10,000
	ILW	76 cm canister	11,700	10,500	11,200
		55 gal drum	3,900	3,500	3,700
	LLW	55 gal drum	1,300	1,000	800
		1.8 x 1.8 x 1.2 m box	15,900	12,300	9,000
Granite	HLW[b]	All HLW canisters	44,900	44,700	45,100
	FRW	76 cm canister	16,700	16,700	17,100
	ILW	76 cm canister	17,700	17,800	18,100
		55 gal drum	5,900	5,900	6,000
	LLW	55 gal drum	2,000	1,900	1,200
		1.8 x 1.8 x 1.2 m box	23,800	22,500	14,900
Shale	HLW[b]	All HLW canisters	28,500	29,300	29,400
	FRW	76 cm canister	12,500	12,500	12,900
	ILW	76 cm canister	14,600	14,900	14,900
		55 gal drum	4,900	5,000	5,000
	LLW	55 gal drum	1,800	1,700	1,100
		1.8 x 1.8 x 1.2 m box	22,000	20,800	13,800
Basalt	HLW[b]	All HLW canisters	34,700	36,600	36,600
	FRW	76 cm canister	17,500	18,300	18,300
	ILW	76 cm canister	19,200	19,300	19,800
		55 gal drum	6,400	6,400	6,600
	LLW	55 gal drum	2,400	2,300	1,500
		1.8 x 1.8 x 1.2 m box	29,300	27,200	17,800

a. All canisters are 3.08 m (10 ft) in length.
b. Due to limiting thermal criteria, HLW canister diameters may vary from 15 to 30 cm. However, since all canisters are assumed to be emplaced in the same configuration, the cost per canister for disposal is the same for all canisters. The cost per kilogram of heavy metal for HLW calculated in Tables 7.5.23 through 7.5.25 is an average for all HLW canister sizes in the first repository.

The shaft and underground construction schedule is shown in Figure 7.5.14. The construction phase for the underground radioactive waste disposal facilities is performed in the following sequence:

- Site preparation and preliminary drilling activities include contractor mobilization and core drilling to verify conditions at the location of the initial shaft. After the preliminary drilling has provided sufficient information about subsurface conditions and the shaft location is established, the shaft sinking contractor initiates operations starting with the canistered waste (CW) shaft.
- The CW shaft dimensions are 4.9 m (16 ft) diameter in excavation, finished to 4.3 m (14 ft) with a 0.3-m (1-ft) thick concrete lining. The shaft has a mine station connecting to the HLW emplacement area at the bottom and an intermediate-level mine station that connects the shaft with the FRW/ILW emplacement area.

This first penetration to the mine uses either the conventional "bench method" for shaft sinking or the "blind hole" drilling method for shaft sinking as described in Section 7.4.11.1.

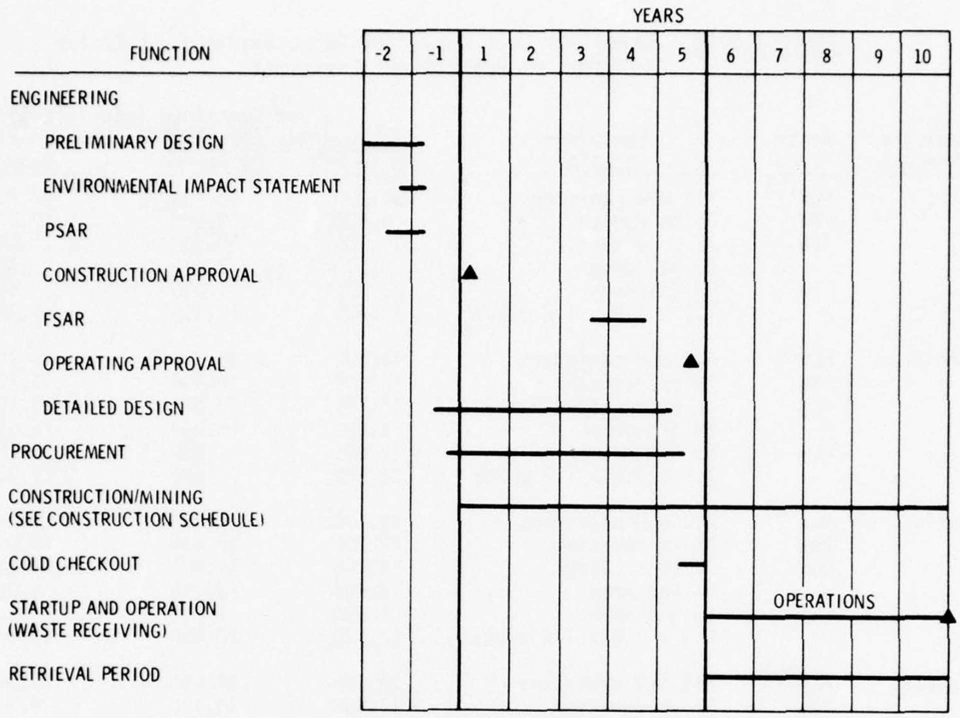


FIGURE 7.5.13. Engineering and Construction Schedule for Reprocessing Fuel Cycle Repositories

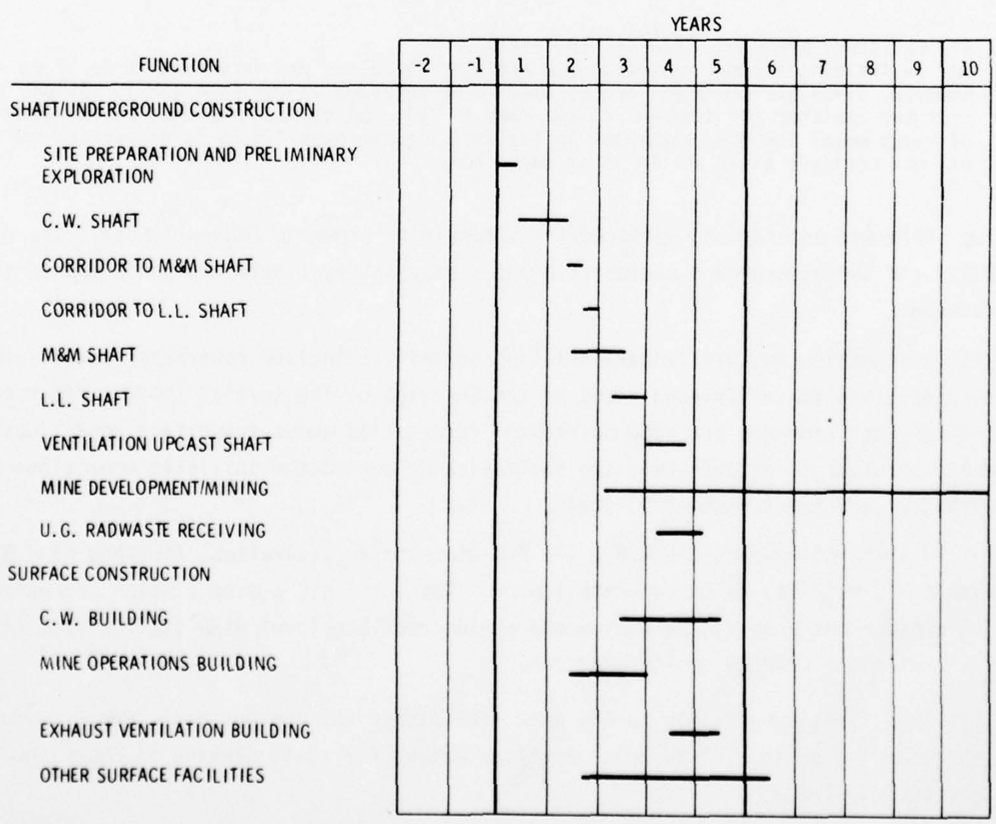


FIGURE 7.5.14. Construction Schedule for Reprocessing Fuel Cycle Repositories

After completion of the CW shaft temporary services and a skip hoist are installed to support underground operation while succeeding shafts are being sunk.

- From the CW shaft FRW/ILW mine station, a corridor is driven toward the men and materials (M&M) shaft. At the end of the corridor (the bottom of the future shaft), an assembly room is excavated where the raise boring bit is assembled. From the surface, a directional hole is drilled to assure a vertical shaft. The drill rods required to upream the shaft are lowered through this hole. The M&M shaft dimensions are 8.2 to 9.5 m (27 to 31 ft) diameter, with 0.3 m (1 ft) minimum of concrete lining. The M&M shaft cannot be upreamed in one stage because of its dimensions. Therefore, a 1.8-m (6-ft) excavation diameter shaft is first raise-bored within the M&M shaft. The enlargement of the excavation and the concrete liner placement is performed by slashing the walls downwards. The muck is removed from the bottom of the shaft through a chute and lifted to the surface by the temporary CW shaft skip hoist.

    A light-weight headframe with a small hoist lowers men and materials during the sinking of the shaft. Following completion of the M&M shaft, the permanent headframe and skip hoist are installed and use of the temporary services in the CW Shaft is discontinued.

- At the granite and basalt repositories a mine production (MP) shaft is constructed with similar techniques and at the same time as the M&M shaft. This shaft is 8.2 to 9.5 m (27 to 31 ft) in diameter and contains skip hoists for removal of mined rock.

- In the next construction sequence, a corridor is driven from the M&M shaft toward the low-level waste (LLW) shaft. At the end of this corridor, an assembly room is excavated to erect the boring bit to raise the LLW shaft. The LLW shaft is bored to 3.7-m (12-ft) diameter and finished to a diameter of 3 m (10 ft) with a 0.3-m (1-ft) concrete lining.

- The ventilation exhaust shaft is constructed in a sequence similar to that for the M&M shaft. This shaft is bored 7.9 to 8.5 m (26 to 28 ft) in diameter and lined with 0.3 m (1 ft) of concrete. No hoist is required.

- From each shaft the subcontractor drives the main mine level corridors to a distance of 60 m (200 ft). This distance is sufficient to erect the mining machinery and begin development of the central underground facilities including those for underground radioactive waste receiving facilities.

- For a repository in salt the mining operations are performed using electrically powered continuous mining units that cut an opening 5.5 m (18 ft) wide x 3.4 m (11 ft) high. On a second pass the mining machine enlarges the opening to 11 m (36 ft) wide by 3.4 m (11 ft) high. Two later passes in the floor produce a corridor or room with dimensions of 11 m (36 ft) x 6.1 m (20 ft). A total of 4 machines passes are required. For these 6.1-m high rooms, the floor cutting involves unbalanced pressures on the mining machine rotors and increased maintenance costs. For the 5.5-m wide rooms, only two passes are required.

For the long rooms, extendible conveyors in each room transfer the mined salt to roofsuspended main line conveyor systems that transport the salt to an underground bin. A reclaim conveyor beneath the storage bins is fed by vibratory feeders that are automatically controlled. The salt is held in a surge bin before being conveyed to skip hoists for removal to the surface.

In the short room system, the constraints on mine geometry require that a loader and truck method that is more flexible be used for the mining operation. Trucks transport the mined salt from the rear of the continuous mining machine to the branch corridor conveyors, which are frequently extended so that truck haulage distances are kept as short as possible. The branch corridor conveyors then transfer the salt to the main line conveyor.

Electrically-operated trucks could be used in the rooms, but the haulage distances are fairly long for trailing cables, and trolley wire systems are not practical for continuous mining systems. Therefore, diesel trucks are used for this operation. Normally, two trucks operate behind one mining unit, but often in restricted areas, only one truck can operate efficiently. On very long hauls three or more trucks may be used.

- Conventional drilling and blasting techniques are used to mine all corridors, rooms, service areas and ventilation corridors for repositories located in shale, basalt and granite. A mining cycle consists of drilling the drift face, loading the holes with explosives, blasting, and after allowing time for fumes to clear, muck removal. Drilling is performed by jumbos equipped with electric hydraulic drills. The drilling crew drills between 60 and 90 5-cm (2-in.) diameter holes per face, depending on the size of each face. The hole length is 3.3 m (11 ft) which allows the breaking of 3 m (10 ft) of ground. When the drilling of a face is completed, the drill crew moves to a second face and commences work while preparations for blasting are being made at the first face.

At the first face drilled, a blasting crew loads the holes with a round of explosives and the round is blasted. While the fumes clear, the blasting crew moves on to another face that already has been drilled.

After the fumes have cleared, a mucking crew begins removing the blasted rock. The mucking crew uses a rubber-tired front-end loader equipped with a 4.6-m^3 (6-yd^3) rock bucket to load the broken rock into 32-MT (35-ton) 17-m^3 (22-yd^3) dump-trucks and when the rooms are being mined, the rock is sent down the branch corridor toward the main corridor system. A grizzly located at the end of the branch corridor (just off to the side of the main corridor) uses a hydraulic breaker to reduce the size of the material. All the broken rock passing through the grizzly is dropped through a vertical opening excavated in the rock. When mining corridors and ventilation drifts, the broken rock is dropped into the nearest chute.

From the control chute, the broken rock is fed into rail cars and transported to a centrally located rock crusher. The crushed rock (maximum 0.2-m (8-in.) material) is fed onto a conveyor located below the crusher discharge. The conveyor carries the crushed rock up a 25% incline slope and discharges it into a storage bin which acts as a surge

control for hoisting operations, located near the M&M shaft. The broken rock is fed onto two short conveyor belts located directly below the storage bin. These belts feed the skip hoist in the men and material shaft.

After rock removal, a fourth crew may be required for bolting and scaling. The bolting crew uses a mobile bolting platform to install rock bolts in the roof and walls in a 1.2-m (4-ft) by 1.2-m pattern. Crews following the rock bolting crews install pipe lines and ventilation tubing, and maintain the roadways. Each crew moves on to another face upon finishing its task. This allows advances on a number of faces simultaneously. This process of multiple heading advance is only applied during the full mining stages.

The construction phase for surface facilities includes:

- Permanent M&M shaft headframes and hoists, and mined materials handling facilities. The mine operations building is also included. Construction of these facilities proceeds concurrently with the underground work.
- Construction of the canistered waste building, low-level waste building and mine exhaust filter building begins upon completion of the corresponding shaft sinking operations and the discontinuance of the use of the CW shaft for temporary mining operations. The construction sequence for these operations is shown on the Construction Schedule (Figure 7.5.14).

7.5.11.2 Construction Labor Requirements

The construction and mining labor requirements are estimated to be:

	Man-Hours, 1000s			
	Salt	Granite	Shale	Basalt
Manual	18,000	37,000	22,000	43,000
Nonmanual	4,000	8,000	4,000	9,000
Total	22,000	45,000	26,000	52,000

These man-hours are distributed over the five-year construction period and the first five years of operation as shown in Figures 7.5.15 through 7.5.18 for repositories in salt, granite, shale, and basalt. The annual construction labor force may be determined on the basis of 2000 man-hours per man-year.

7.5.11.3 Distribution Between Onsite and Offsite Costs

Onsite costs are those for all construction, materials and services provided at the site of the repository while offsite costs are those for all services provided, equipment fabricated and/or assembled, and material purchased offsite of the repository. The distribution of total costs in these categories for the U & Pu recycle repositories is as shown below. The distribution of costs for the U-only recycle repositories is approximately the same.

	Facility Costs (mid-1978 $1000s)			
	Salt	Granite	Shale	Basalt
Onsite	310,000	540,000	330,000	610,000
Offsite	700,000	1,210,000	750,000	1,365,000
Total	1,010,000	1,750,000	1,080,000	1,975,000

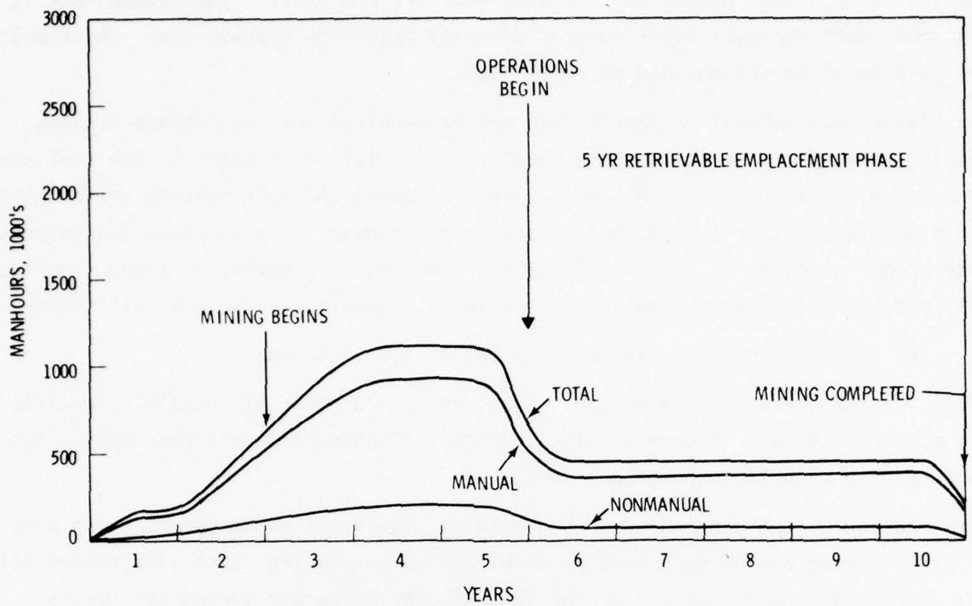


FIGURE 7.5.15. Construction Manpower Requirements - Reprocessing Fuel Cycle Repository in Salt

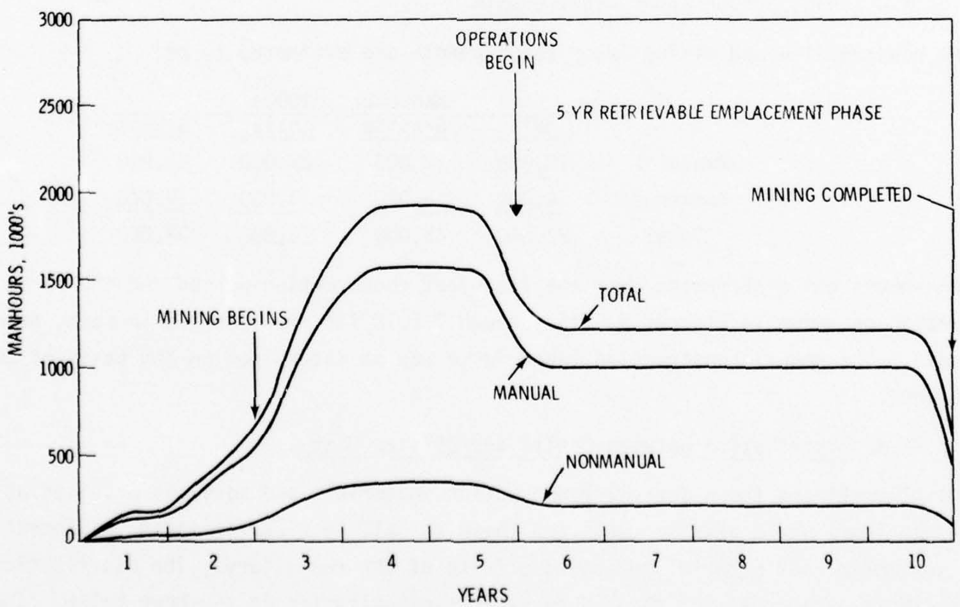


FIGURE 7.5.16. Construction Manpower Requirements - Reprocessing Fuel Cycle Repository in Granite

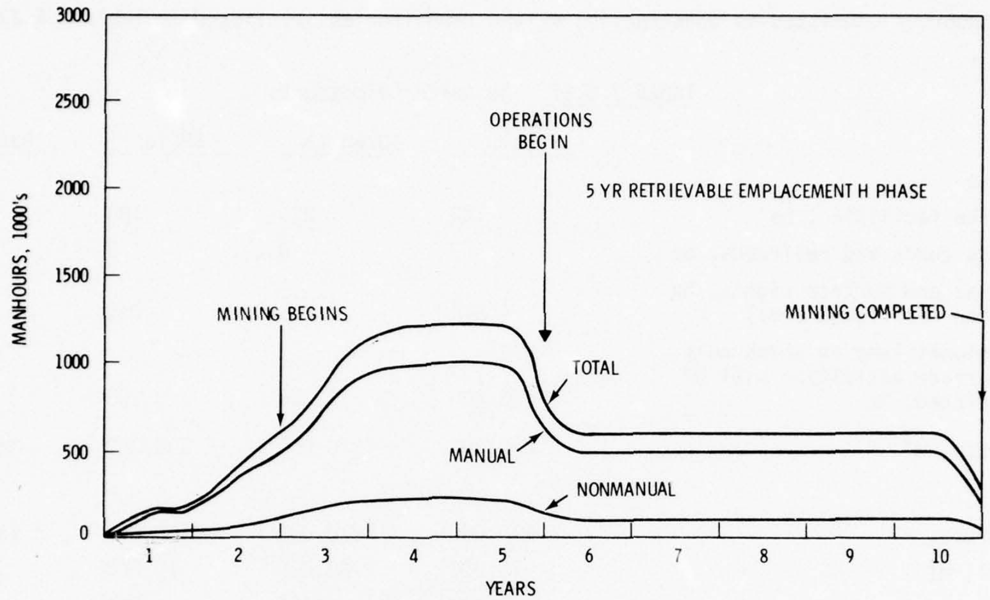


FIGURE 7.5.17. Construction Manpower Requirements - Reprocessing Fuel Cycle Repository in Shale

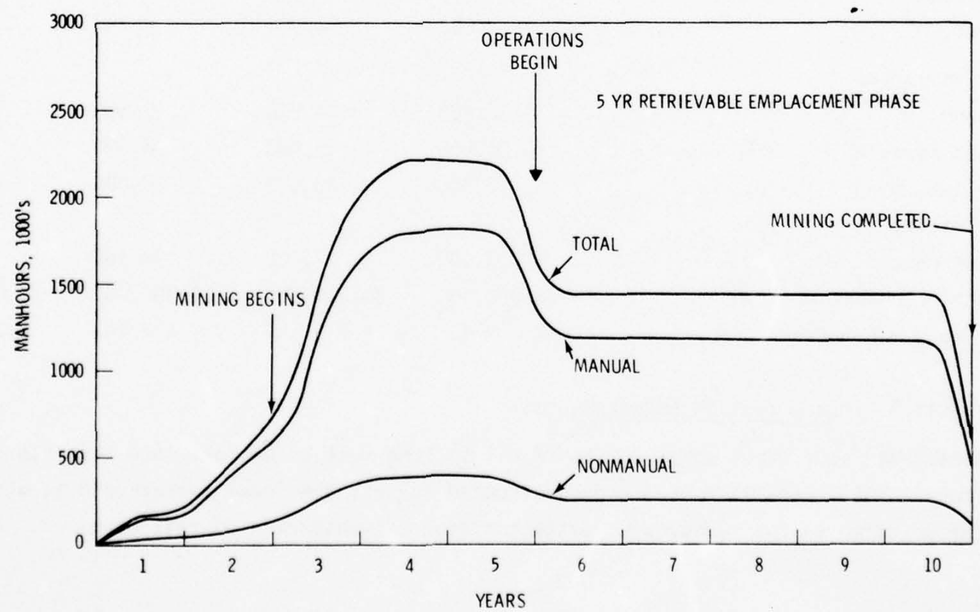


FIGURE 7.4.18. Construction Manpower Requirements - Reprocessing Fuel Cycle Repository in Basalt

7.5.11.4 Resources Committed

Resources committed to construction of the repositories are listed in Table 7.5.27.

TABLE 7.5.27. Resource Commitments

	Salt	Granite	Shale	Basalt
Land Use				
Surface facilities, ha	180	220	180	220
Access roads and railroads, ha	8	8	8	8
Mineral and surface rights, ha (fenced restricted area)	810	810	810	810
Additional land on which only subsurface activities will be restricted, ha	3,200	3,200	3,200	3,200
Water Use, m^3	270,000	510,000	290,000	450,000
Materials				
Concrete, m^3	110,000	210,000	120,000	190,000
Steel, MT	18,000	33,000	19,000	30,000
Copper, MT	240	470	260	420
Zinc, MT	62	120	67	110
Aluminum, MT	46	90	50	77
Lumber, m^3	2,600	4,900	2,800	4,400
Energy Resources				
Propane, m^3	2,400	4,500	2,600	4,000
Diesel fuel, m^3	24,000	45,000	26,000	40,000
Gasoline, m^3	18,000	33,000	19,000	30,000
Electricity				
Peak demand, kW	3,900	7,300	4,100	6,600
Total consumption, kWh	16,000,000	30,000,000	17,000,000	27,000,000
Manpower, man-hours	2.2×10^7	4.5×10^7	2.6×10^7	5.2×10^7

7.5.11.5 Transportation Requirements

A railroad spur track about 3.2 km (2 miles) long must be brought into the site of the repository. This track is routed and constructed to suit the local terrain and is ultimately used for bringing in the radioactive waste containers transported on rail cars.

REFERENCES FOR SECTION 7.5

1. Union Carbide Corporation, Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Y/OWI/TM-44, Office of Waste Isolation, Union Carbide Corporation, Nuclear Division, Oak Ridge, TN, 1978.

2. Union Carbide Corporation, Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations, Y/OWI/TM-36, Office of Waste Isolation, Union Carbide Corporation, Nuclear Division, Oak Ridge, TN, 1978.

3. Environmental Aspects of Commercial Waste Management, DOE/ET-0029, p. 7.5.45, Department of Energy, Washington, DC, in press.

7.6 PHYSICAL PROTECTION AND SAFEGUARD REQUIREMENTS FOR REPOSITORIES

AD-A078 258 DEPARTMENT OF ENERGY WASHINGTON DC ASSISTANT SECRETA--ETC F/G 18/7
TECHNOLOGY FOR COMMERCIAL RADIOACTIVE WASTE MANAGEMENT. VOLUME --ETC(U)
MAY 79

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7.6 PHYSICAL PROTECTION AND SAFEGUARD REQUIREMENTS FOR GEOLOGIC REPOSITORIES

Geologic repository facilities will require physical protection and safeguards to prevent the theft of plutonium-bearing wastes or highly radioactive waste materials or to prevent the sabotage of the facility. Two waste materials proposed for geologic disposal would be potential targets for theft for weapons material: spent fuel and solidified high-level waste containing discarded plutonium from the uranium-only recycle option. The quantity and concentration of plutonium in these two wastes makes them potentially more attractive for theft than the other waste forms. The safeguards provisions for the facility handling these waste forms must be similar to those required for facilities processing strategic quantities of plutonium. These safeguards requirements, summarized below, will adequately protect the other wastes, which are much less attractive as targets for theft or sabotage.

Safeguards of the surface facilities at the repository site would receive the principal emphasis. After placement in the geologic repository the waste would be essentially unavailable for theft or dispersion through sabotage. A successful intrusion and theft would be unlikely because of the operational controls over entry and the physical security provided at the access points in the surface facility.

The physical protection requirements of nuclear facilities and materials are specified in the Code of Federal Regulations (10 CFR Parts 50, 70 and 73). The principal requirements are the use of armed guards, physical and procedural access controls, intrusion detection devices, secure communications systems, and contingency planning for assistance in emergencies. Equipment items, systems, devices, or materials whose failure, destruction, or release could directly or indirectly endanger the public health and safety by exposure to radiation are considered "vital," and such equipment is subject to additional specific protective measures. The wastes containing substantial quantities of plutonium would be classified as vital material and therefore would be located in controlled access areas protected by two physical barriers.

There would be no significant differences expected in the safeguards risks and the physical protection requirements prior to and subsequent to disposal among acceptable geologic repositories in salt, granite, shale or basalt.

APPENDIX 7A

CONSTRUCTION AND MINING COSTS ESTIMATES AND MINING EXPENDITURE SCHEDULES FOR GEOLOGIC REPOSITORIES ONCE-THROUGH FUEL CYCLE

APPENDIX 7A

CONSTRUCTION AND MINING COST ESTIMATES AND MINING EXPENDITURE SCHEDULES FOR GEOLOGIC REPOSITORIES ONCE-THROUGH FUEL CYCLE

APPENDIX 7A

CONSTRUCTION AND MINING COST ESTIMATES AND MINING EXPENDITURE SCHEDULES FOR GEOLOGIC REPOSITORIES - ONCE-THROUGH FUEL CYCLE

Detailed construction and mining cost estimates for geologic repositories in salt, granite, shale and basalt are given in the following tables for both accelerated and continuous mining options for the once-through fuel cycle. The accelerated mining case assumes that the repository is fully mined out in ten years with all mined rock being stored on the surface during this period. The continuous mining alternative assumes that mining occurs throughout the repository operating life at a rate commensurate with waste receipt rates.

The mining expenditure schedules for the above media for the continuous mining option are also shown. Expenditure schedules for facility construction and for accelerated mining are given in the text in Section 7.4.10.1.

TABLE 7.A.1. Capital Cost Estimate for Geologic Isolation in Salt - Once-Through Cycle - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	24,380	27,220	51,600
Major equipment	43,350	3,990	47,340
Bulk material	29,040	16,410	45,450
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	12,730	4,250	16,980
Shafts and lining	41,960	25,600	67,560
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,930
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	49,890	---	49,890
Mine construction	102,980	117,020	220,000
Backfilling			
Mine backfilling	32,390	36,500	68,890
Shaft sealing	70	80	150
Total Field Costs	352,000	242,000	594,000
Architect-Engineer Services			45,000
Owner's Costs			70,000
Contingency			174,000
TOTAL FACILITY COST			883,000

TABLE 7.A.2. Capital Cost Estimate for Geologic Isolation in Salt - Once-Through Cycle - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	24,380	27,220	51,600
Major equipment	42,340	3,910	46,250
Bulk material	28,370	16,410	44,780
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	10,560	3,520	14,080
Shafts and lining	41,960	22,600	64,560
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	49,890	---	49,890
Mine construction	103,290	116,790	220,080
Backfilling			
Mine backfilling	23,010	25,590	48,600
Shaft sealing	70	80	150
Total Field Costs	339,000	227,000	566,000
Architect-Engineer Services			45,000
Owner's Costs			68,000
Contingency			169,000
TOTAL FACILITY COST			848,000

TABLE 7.A.3. Capital Cost Estimate Geologic Isolation in Granite - Once-Through Cycle - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	29,380	33,640	63,020
Major equipment	46,350	4,570	50,920
Bulk material	29,040	16,410	45,450
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	21,400	7,300	28,700
Shafts and lining	51,500	27,600	79,100
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	477,830	536,940	1,014,770
Backfilling			
Mine backfilling	85,500	96,500	182,000
Shaft sealing	90	110	200
Total Field Costs	821,000	734,000	1,555,000
Architect-Engineer Services			50,000
Owner's Costs			185,000
Contingency			453,000
TOTAL FACILITY COST			2,243,000

TABLE 7.A.4. Capital Cost Estimate for Geologic Isolation in Granite - Once-Through Cycle - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	29,380	33,640	63,020
Major equipment	45,340	4,490	48,360
Bulk material	28,370	16,410	44,780
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	21,400	7,300	28,700
Shafts and lining	41,900	22,500	64,400
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	478,690	536,670	1,015,360
Backfilling			
Mine backfilling	68,000	78,000	146,000
Shaft sealing	90	110	200
Total Field Costs	793,000	710,000	1,503,000
Architect-Engineer Services			47,000
Owner's Costs			180,000
Contingency			434,000
TOTAL FACILITY COST			2,164,000

TABLE 7.A.5. Capital Cost Estimate for Geologic Isolation in Shale - Once-Through Cycle - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	26,180	29,230	55,410
Major equipment	43,550	4,980	48,530
Bulk material	29,040	16,410	45,450
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	13,800	4,900	18,700
Shafts and lining	26,950	14,500	41,450
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	51,700	---	51,700
Mine construction	188,480	211,940	400,420
Backfilling			
Mine backfilling	39,000	45,000	84,000
Shaft sealing	90	110	200
Total Field Costs	434,000	338,000	772,000
Architect-Engineer Services			42,000
Owner's Costs			92,000
Contingency			226,000
TOTAL FACILITY COST			1,132,000

TABLE 7.A.6. Capital Cost Estimate for Geologic Isolation in Shale - Once-Through Cycle - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	26,180	29,230	55,410
Major equipment	42,540	4,900	47,440
Bulk material	28,370	16,410	44,780
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	13,800	4,900	18,700
Shafts and lining	23,800	12,850	36,650
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,440	3,430
Mining			
Major equipment	51,700	---	51,700
Mine construction	188,390	211,720	400,110
Backfilling			
Mine backfilling	32,000	36,000	68,000
Shaft sealing	90	110	200
Total Field Costs	422,000	327,000	749,000
Architect-Engineer Services			41,000
Owner's Costs			90,000
Contingency			217,000
TOTAL FACILITY COST			1,097,000

TABLE 7.A.7. Capital Cost Estimate for Geologic Isolation in Basalt - Once-Through Cycle - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	29,380	33,640	63,020
Major equipment	46,350	4,570	50,920
Bulk material	29,040	16,410	45,450
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	24,500	8,300	32,800
Shafts and lining	58,100	31,400	89,500
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	582,330	655,640	1,237,970
Backfilling			
Mine backfilling	102,300	116,000	218,300
Shaft sealing	90	110	200
Total Field Costs	952,000	877,000	1,829,000
Architect-Engineer Services			53,000
Owner's Costs			218,000
Contingency			534,000
TOTAL FACILITY COST			2,634,000

TABLE 7.A.8. Capital Cost Estimate for Geologic Isolation in Basalt - Once-Through Cycle - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	29,380	33,640	63,020
Major equipment	45,340	4,490	48,360
Bulk material	28,370	26,410	44,780
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	24,500	8,300	32,800
Shafts and lining	47,800	25,700	73,500
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	581,690	655,870	1,237,560
Backfilling			
Mine backfilling	82,000	92,600	174,600
Shaft sealing	90	110	200
Total Field Costs	919,000	848,000	1,767,000
Architect-Engineer Services			50,000
Owner's Costs			210,000
Contingency			507,000
TOTAL FACILITY COST			2,534,000

TABLE 7.A.9. Expenditure Schedule for Continuous Mining - Salt Repository - Once-Through Cycle

Time, years	Material Percent Expended	Material Cumulative	Labor Percent Expended	Labor Cumulative
1	--	--	--	--
2	1.4	1.4	--	--
3	8.2	9.6	3.0	3.0
4	8.2	17.8	6.0	9.0
5	5.4	23.2	6.0	15.0
6	5.4	28.6	6.0	21.0
7	5.4	34.0	6.0	27.0
8	5.4	39.4	6.0	33.0
9	5.4	44.8	6.0	39.0
10	5.4	50.2	6.0	45.0
11	5.4	55.6	6.0	51.0
12	5.4	61.0	6.0	57.0
13	5.4	66.4	6.0	63.0
14	5.4	71.8	6.0	69.0
15	5.4	77.2	6.0	75.0
16	5.4	82.6	6.0	81.0
17	5.4	88.0	6.0	87.0
18	5.4	93.4	6.0	93.0
19	5.4	98.8	6.0	99.0
20	1.2	100.0	1.0	100.0

TABLE 7.A.10. Expenditure Schedule for Continuous Mining - Granite Repository - Once-Through Cycle

Time, years	Material Percent Expended	Material Cumulative	Labor Percent Expended	Labor Cumulative
1	--	--	--	--
2	1.8	1.8	--	--
3	5.4	7.2	1.3	1.2
4	5.4	12.6	3.9	5.2
5	3.6	16.2	3.9	9.1
6	3.6	19.8	3.9	13.0
7	3.6	23.4	3.9	16.9
8	3.6	27.0	3.9	20.8
9	3.6	30.6	3.9	24.7
10	3.6	34.2	3.9	28.6
11	3.6	37.8	3.9	32.5
12	3.6	41.4	3.9	36.4
13	3.6	45.0	3.9	40.3
14	3.6	48.6	3.9	44.2
15	3.6	52.2	3.9	48.1
16	3.6	55.8	3.9	52.0
17	3.6	59.4	3.9	55.9
18	3.6	63.0	3.9	59.8
19	3.6	66.6	3.9	63.7
20	3.6	70.2	3.9	67.6
21	3.6	73.8	3.9	71.5
22	3.6	77.4	3.9	75.4
23	3.6	81.0	3.9	79.3
24	3.6	84.6	3.9	83.2
25	3.6	88.2	3.9	87.1
26	3.6	91.8	3.9	91.0
27	3.6	95.4	3.9	94.9
28	3.6	99.0	3.9	98.8
29	1.0	100.0	1.2	100.0

TABLE 7.A.11. Expenditure Schedule for Continuous Mining - Shale Repository - Once-Through Cycle

Time, years	Material: Percent Expended	Material: Cumulative	Labor: Percent Expended	Labor: Cumulative
1	--	--	--	--
2	1.1	1.1	--	--
3	7.3	8.4	2.7	2.7
4	7.3	15.7	5.3	8.0
5	4.9	20.6	5.3	13.3
6	4.9	25.5	5.3	18.6
7	4.9	30.4	5.3	23.9
8	4.9	35.3	5.3	29.2
9	4.9	40.2	5.3	34.5
10	4.9	45.1	5.3	39.8
11	4.9	50.0	5.3	45.1
12	4.9	54.9	5.3	50.4
13	4.9	59.8	5.3	55.7
14	4.9	64.7	5.3	61.0
15	4.9	69.6	5.3	66.3
16	4.9	74.5	5.3	71.6
17	4.9	79.4	5.3	76.9
18	4.9	84.3	5.3	82.2
19	4.9	89.2	5.3	87.5
20	4.9	94.1	5.3	92.8
21	4.9	99.0	5.3	98.1
22	1.0	100.0	1.9	100.0

TABLE 7.A.12. Expenditure Schedule for Continuous Mining - Basalt Repository - Once-Through Cycle

Time, years	Material: Percent Expended	Material: Cumulative	Labor: Percent Expended	Labor: Cumulative
1	--	--	--	--
2	1.8	1.8	--	--
3	5.4	7.2	1.3	1.3
4	5.4	12.6	3.9	5.2
5	3.6	16.2	3.9	9.1
6	3.6	19.8	3.9	13.0
7	3.6	23.4	3.9	16.9
8	3.6	27.0	3.9	20.8
9	3.6	30.6	3.9	24.7
10	3.6	24.2	3.9	28.6
11	3.6	37.8	3.9	32.5
12	3.6	41.4	3.9	36.4
13	3.6	45.0	3.9	40.3
14	3.6	48.6	3.9	44.2
15	3.6	52.2	3.9	48.1
16	3.6	55.8	3.9	52.0
17	3.6	59.4	3.9	55.9
18	3.6	63.0	3.9	59.8
19	3.6	66.6	3.9	63.7
20	3.6	70.2	3.9	67.6
21	3.6	73.8	3.9	71.5
22	3.6	77.4	3.9	75.4
23	3.6	81.0	3.9	79.3
24	3.6	84.6	3.9	83.2
25	3.6	88.2	3.9	87.1
26	3.6	91.8	3.9	91.0
27	3.6	95.4	3.9	94.9
28	3.6	99.0	3.9	98.8
29	1.0	100.0	1.2	100.0

APPENDIX 7B

CONSTRUCTION AND MINING COST ESTIMATES AND MINING EXPENDITURE SCHEDULES FOR GEOLOGIC REPOSITORIES RECYCLE FUEL CYCLES

APPENDIX 7B

CONSTRUCTION AND MINING COST ESTIMATES AND MINING EXPENDITURE SCHEDULES FOR GEOLOGIC REPOSITORIES - RECYCLE FUEL CYCLES

APPENDIX 7B

CONSTRUCTION AND MINING COST ESTIMATES AND MINING EXPENDITURE SCHEDULES FOR GEOLOGIC REPOSITORIES - RECYCLE FUEL CYCLES

Detailed construction and mining cost estimates for geologic repositories in salt, granite, shale and basalt are given for accelerated and continuous mining alternatives for three recycle fuel cycles: U-Only Recycle - Pu to HLW; U-Only Recycle - Pu Stored and U and Pu Recycle. The mining expenditure schedules for the continuous mining option for the above media and fuel cycles are also shown. Expenditure schedules for facility construction and accelerated mining are given in the text in Section 7.5.10.1.

TABLE 7.B.1. Capital Cost Estimate for Geologic Isolation in Salt - U-Only Recycle; Pu to HLW - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	29,690	33,600	63,290
Major equipment	45,690	4,700	50,390
Bulk material	30,750	17,100	47,850
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	13,090	4,370	17,460
Shafts and lining	44,860	27,160	72,020
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	49,890	---	49,890
Mine construction	103,390	116,560	219,950
Backfilling			
Mine backfilling	32,350	36,500	68,850
Shaft sealing	80	80	160
Total Field Costs	365,000	251,000	616,000
Architect-Engineer Services			50,000
Owner's Costs			74,000
Contingency			189,000
TOTAL FACILITY COST			929,000

TABLE 7.B.2. Capital Cost Estimate for Geologic Isolation in Salt - U-Only Recycle; Pu Stored - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	32,300	36,660	68,960
Major equipment	47,690	4,700	52,390
Bulk material	30,750	17,100	47,850
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	13,090	4,370	17,460
Shafts and lining	44,860	27,160	72,020
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	49,890	---	49,890
Mine construction	119,130	134,150	253,280
Backfilling			
Mine backfilling	38,000	42,850	80,850
Shaft sealing	80	80	160
Total Field Costs	391,000	278,000	669,000
Architect-Engineer Services			51,000
Owner's Costs			80,000
Contingency			200,000
TOTAL FACILITY COST			1,000,000

TABLE 7.B.3. Capital Cost Estimate for Geologic Isolation in Salt - U and Pu Recycle - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	32,300	36,660	68,960
Major equipment	47,690	4,700	52,390
Bulk material	30,750	17,700	47,850
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	13,090	4,370	17,460
Shafts and lining	44,860	27,160	72,020
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	49,890	---	49,890
Mine construction	121,940	138,000	259,940
Backfilling			
Mine backfilling	37,190	42,000	79,190
Shaft sealing	80	80	160
Total Field Costs	393,000	281,000	674,000
Architect-Engineer Services			51,000
Owner's Costs			80,000
Contingency			204,000
TOTAL FACILITY COST			1,009,000

TABLE 7.B.4. Capital Cost Estimate for Geologic Isolation in Salt - U-Only Recycle; Pu to HLW - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	29,690	33,600	63,290
Major equipment	44,680	4,620	49,300
Bulk material	29,940	17,100	47,040
Site development	7,490	4,480	12,170
Shafts and Hoists			
Major equipment	10,920	3,640	14,560
Shafts and lining	44,860	24,160	69,020
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	49,890	---	49,890
Mine construction	102,800	117,280	220,080
Backfilling			
Mine backfilling	23,010	25,640	48,650
Shaft sealing	80	80	160
Total Field Costs	351,000	237,000	588,000
Architect-Engineer Services			49,000
Owner's Costs			71,000
Contingency			176,000
TOTAL FACILITY COST			884,000

TABLE 7.B.5. Capital Cost Estimate for Geologic Isolation in Salt - U-Only Recycle; Pu Stored - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	32,300	36,660	68,960
Major equipment	46,680	4,620	51,300
Bulk material	29,940	17,100	47,040
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	10,920	3,640	14,560
Shafts and lining	44,860	24,160	69,020
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk Material	1,960	1,470	3,430
Mining			
Major equipment	49,890	---	49,890
Mine construction	119,260	134,160	253,420
Backfilling			
Mine backfilling	16,950	18,700	35,650
Shaft sealing	70	80	150
Total Field Costs	366,000	250,000	616,000
Architect-Engineer Services			50,000
Owner's Costs			74,000
Contingency			189,000
TOTAL FACILITY COST			929,000

TABLE 7.B.6. Capital Cost Estimate for Geologic Isolation in Salt - U and Pu Recycle - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	32,300	36,660	68,960
Major equipment	46,680	4,620	51,300
Bulk material	29,940	17,100	47,040
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	10,920	3,640	14,560
Shafts and lining	44,860	24,160	69,020
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	49,890	---	49,890
Mine construction	122,060	138,030	260,090
Backfilling			
Mine backfilling	17,140	19,830	36,970
Shaft sealing	80	80	160
Total Field Costs	369,000	255,000	624,000
Architect-Engineer Services			50,000
Owner's Costs			75,000
Contingency			188,000
TOTAL FACILITY COST			937,000

TABLE 7.B.7. Capital Cost Estimate for Geologic Isolation in Granite - U-Only Recycle; Pu to HLW - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	33,690	38,600	72,290
Major equipment	48,690	4,700	53,390
Bulk material	30,750	17,100	47,850
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	22,500	7,600	30,100
Shafts and lining	55,200	29,600	84,800
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	320,670	359,560	680,230
Backfilling			
Mine backfilling	57,500	64,800	122,300
Shaft sealing	90	110	200
Total Field Costs	649,000	533,000	1,182,000
Architect-Engineer Services			55,000
Owner's Costs			141,000
Contingency			341,000
TOTAL FACILITY COST			1,719,000

TABLE 7.B.8. Capital Cost Estimate for Geologic Isolation in Granite - U-Only Recycle; Pu Stored - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	34,300	38,060	72,360
Major equipment	48,690	5,300	53,990
Bulk material	30,750	17,100	47,850
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	22,500	7,600	30,100
Shafts and lining	55,200	29,600	84,800
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	321,060	359,500	680,560
Backfilling			
Mine backfilling	57,500	64,800	122,300
Shaft sealing	90	110	200
Total Field Costs	650,000	533,000	1,183,000
Architect-Engineer Services			56,000
Owner's Costs			142,000
Contingency			347,000
TOTAL FACILITY COST			1,728,000

TABLE 7.B.9. *Capital Cost Estimate for Geologic Isolation in Granite - U and Pu Recycle - Accelerated Mining*

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	34,300	38,060	72,360
Major equipment	48,690	5,300	53,990
Bulk material	30,750	17,100	47,850
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	22,500	7,600	30,100
Shafts and lining	55,200	29,600	84,800
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	326,060	366,400	692,460
Backfilling			
Mine backfilling	58,500	65,900	124,400
Shaft sealing	90	110	200
Total Field Costs	656,000	541,000	1,197,000
Architect-Engineer Services			56,000
Owner's Costs			143,000
Contingency			349,000
TOTAL FACILITY COST			1,745,000

TABLE 7.B.10. Capital Cost Estimate for Geologic Isolation in Granite - U-Only Recycle; Pu to HLW - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	33,690	38,600	72,290
Major equipment	47,680	4,620	52,300
Bulk material	29,940	17,100	47,040
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	22,500	7,600	30,100
Shafts and lining	45,600	24,500	70,100
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	320,670	359,590	680,260
Backfilling			
Mine backfilling	46,000	52,000	98,000
Shaft sealing	90	110	200
Total Field Costs	626,000	515,000	1,141,000
Architect-Engineer Services			52,000
Owner's Costs			136,000
Contingency			328,000
TOTAL FACILITY COST			1,657,000

TABLE 7.B.11. Capital Cost Estimate for Geologic Isolation in Granite - U-Only Recycle; Pu Stored - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	34,300	38,060	72,360
Major equipment	47,680	5,220	52,900
Bulk material	29,940	17,100	47,040
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	22,500	7,600	30,100
Shafts and lining	45,600	24,500	70,100
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	321,060	369,530	680,590
Backfilling			
Mine backfilling	46,000	52,000	98,000
Shaft sealing	90	110	200
Total Field Costs	627,000	515,000	1,142,000
Architect-Engineer Services			53,000
Owner's Costs			136,000
Contingency			334,000
TOTAL FACILITY COST			1,665,000

TABLE 7.B.12. Capital Cost Estimate for Geologic Isolation in Granite - U and Pu Recycle - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	34,300	38,060	72,360
Major equipment	47,680	5,220	52,900
Bulk material	29,940	17,100	47,040
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	22,500	7,600	30,100
Shafts and lining	45,600	24,500	70,100
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	326,060	366,530	692,590
Backfilling			
Mine backfilling	47,000	53,000	100,000
Shaft sealing	90	110	200
Total Field Costs	633,000	523,000	1,156,000
Architect-Engineer Services			53,000
Owner's Costs			138,000
Contingency			336,000
TOTAL FACILITY COST			1,683,000

TABLE 7.B.13. Capital Cost Estimate for Geologic Isolation in Shale - U-Only Recycle; Pu to HLW - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	31,460	34,100	65,560
Major equipment	46,920	4,200	51,120
Bulk material	30,750	17,100	47,850
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	15,300	5,400	20,700
Shafts and lining	30,550	16,450	47,000
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	51,700	---	51,700
Mine construction	160,020	178,710	338,730
Backfilling			
Mine backfilling	33,000	38,000	71,000
Shaft sealing	90	110	200
Total Field Costs	415,000	305,000	720,000
Architect-Engineer Services			46,000
Owner's Costs			86,000
Contingency			218,000
TOTAL FACILITY COST			1,070,000

TABLE 7.B.14. Capital Cost Estimate for Geologic Isolation in Shale - U-Only Recycle; Pu Stored - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	31,300	34,660	65,960
Major equipment	46,690	4,700	51,390
Bulk material	30,750	17,100	47,850
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	15,300	5,400	20,700
Shafts and lining	30,550	16,450	47,000
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	51,700	---	51,700
Mine construction	159,410	179,650	339,060
Backfilling			
Mine backfilling	33,000	38,000	71,000
Shaft sealing	90	110	200
Total Field Costs	414,000	307,000	721,000
Architect-Engineer Services			47,000
Owner's Costs			87,000
Contingency			224,000
TOTAL FACILITY COST			1,079,000

TABLE 7.B.15. Capital Cost Estimate for Geologic Isolation in Shale - U and Pu Recycle - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	31,300	34,660	65,960
Major equipment	41,690	4,700	51,390
Bulk material	30,750	17,100	47,850
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	15,300	5,400	20,700
Shafts and lining	30,550	16,450	47,000
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	51,700	---	51,700
Mine construction	159,410	179,650	339,060
Backfilling			
Mine backfilling	33,000	38,000	71,000
Shaft sealing	90	110	200
Total Field Costs	414,000	307,000	721,000
Architect-Engineer Services			47,000
Owner's Costs			87,000
Contingency			224,000
TOTAL FACILITY COST			1,079,000

TABLE 7.B.16. Capital Cost Estimate for Geologic Isolation in Shale - U-Only Recycle; Pu to HLW - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	31,460	34,100	65,560
Major equipment	45,910	4,120	50,030
Bulk material	29,940	17,100	47,040
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	15,300	5,400	20,700
Shafts and lining	27,400	14,800	42,200
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	51,700	---	51,700
Mine construction	159,070	179,490	338,560
Backfilling			
Mine backfilling	27,000	30,000	57,000
Shaft sealing	90	110	200
Total Field Costs	403,000	296,000	699,000
Architect-Engineer Services			46,000
Owner's Costs			84,000
Contingency			205,000
TOTAL FACILITY COST			1,034,000

TABLE 7.B.17. Capital Cost Estimate for Geologic Isolation in Shale - U-Only Recycle; Pu Stored - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	31,300	34,660	65,960
Major equipment	45,680	4,620	50,300
Bulk material	29,940	17,100	47,040
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	15,300	5,400	20,700
Shafts and lining	27,400	14,800	42,200
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	51,700	---	51,700
Mine construction	159,460	179,430	338,890
Backfilling			
Mine backfilling	27,000	30,000	57,000
Shaft sealing	90	110	200
Total Field Costs	403,000	297,000	700,000
Architect-Engineer Services			47,000
Owner's Costs			84,000
Contingency			212,000
TOTAL FACILITY COST			1,043,000

TABLE 7.B.18. Capital Cost Estimate for Geologic Isolation in Shale - U and Pu Recycle - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	31,300	34,660	65,960
Major equipment	45,680	4,620	50,300
Bulk material	29,940	17,100	47,040
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	15,300	5,400	20,700
Shafts and lining	27,400	14,800	42,200
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	51,700	---	51,700
Mine construction	159,460	179,430	338,890
Backfilling			
Mine backfilling	27,000	30,000	57,000
Shaft sealing	90	110	200
Total Field Costs	403,000	297,000	700,000
Architect-Engineer Services			47,000
Owner's Costs			84,000
Contingency			212,000
TOTAL FACILITY COST			1,043,000

TABLE 7.B.19. Capital Cost Estimate for Geologic Isolation in Basalt - U-Only Recycle; Pu to HLW - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	33,690	35,600	72,290
Major equipment	48,690	4,700	53,390
Bulk material	30,750	17,100	47,850
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	25,700	8,700	34,400
Shafts and lining	62,300	33,700	96,000
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	382,470	430,260	812,730
Backfilling			
Mine backfilling	67,400	75,900	143,300
Shaft sealing	90	110	200
Total Field Costs	731,000	620,000	1,351,000
Architect-Engineer Services			58,000
Owner's Costs			161,000
Contingency			389,000
TOTAL FACILITY COST			1,959,000

TABLE 7.B.20. Capital Cost Estimate for Geologic Isolation in Basalt - U-Only Recycle; Pu Stored - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	34,300	38,060	72,360
Major equipment	48,690	5,300	53,990
Bulk material	30,750	17,100	47,850
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	25,700	8,700	34,400
Shafts and lining	62,300	33,700	96,000
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	369,960	417,500	787,460
Backfilling			
Mine backfilling	65,300	73,600	138,900
Shaft sealing	90	110	200
Total Field Costs	717,000	605,000	1,322,000
Architect-Engineer Services			59,000
Owner's Costs			158,000
Contingency			394,000
TOTAL FACILITY COST			1,933,000

TABLE 7.B.21. Capital Cost Estimate for Geologic Isolation in Basalt - U and Pu Recycle - Accelerated Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	33,300	38,060	72,360
Major equipment	48,690	5,300	53,990
Bulk material	30,750	17,100	47,850
Site development	7,570	4,730	12,300
Shafts and Hoists			
Major equipment	25,700	8,700	34,400
Shafts and lining	62,300	33,700	96,000
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	381,860	430,200	812,060
Backfilling			
Mine backfilling	67,400	75,900	143,300
Shaft sealing	90	110	200
Total Field Costs	731,000	620,400	1,351,000
Architect-Engineer Services			59,000
Owner's Costs			161,000
Contingency			397,000
TOTAL FACILITY COST			1,968,000

TABLE 7.B.22. Capital Cost Estimate for Geologic Isolation in Basalt - U-Only Recycle; Pu to HLW - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	33,690	35,600	72,290
Major equipment	47,680	4,620	52,300
Bulk material	29,940	17,100	47,040
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	25,700	8,700	34,400
Shafts and lining	52,000	28,000	80,000
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	382,170	430,290	812,460
Backfilling			
Mine backfilling	53,900	60,700	114,600
Shaft sealing	90	110	200
Total Field Costs	705,000	599,000	1,304,000
Architect-Engineer Services			55,000
Owner's Costs			155,000
Contingency			373,000
TOTAL FACILITY COST			1,887,000

TABLE 7.B.23. Capital Cost Estimate for Geologic Isolation in Basalt - U-Only Recycle; Pu Stored - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	34,300	38,060	72,360
Major equipment	48,680	5,220	52,900
Bulk material	29,940	17,100	47,040
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	25,700	8,700	34,400
Shafts and lining	52,000	28,000	80,000
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	370,260	417,030	787,290
Backfilling			
Mine backfilling	52,200	58,900	111,100
Shaft sealing	90	110	200
Total Field Costs	692,000	584,000	1,276,000
Architect-Engineer Services			56,000
Owner's Costs			152,000
Contingency			377,000
TOTAL FACILITY COST			1,861,000

TABLE 7.B.24. Capital Cost Estimate for Geologic Isolation in Basalt U and Pu Recycle - Continuous Mining

	Mid-1976 Costs, $1000s		
	Material	Labor	Total
Surface Facilities			
Buildings and structures	34,300	38,060	72,360
Major equipment	47,680	5,220	52,900
Bulk material	29,940	17,100	47,040
Site development	7,490	4,680	12,170
Shafts and Hoists			
Major equipment	25,700	8,700	34,400
Shafts and lining	52,000	28,000	80,000
Underground Facilities			
Excavations and structures	2,510	4,510	7,020
Major equipment	3,170	220	3,390
Bulk material	1,960	1,470	3,430
Mining			
Major equipment	64,700	---	64,700
Mine construction	381,560	431,230	812,790
Backfilling			
Mine backfilling	53,900	60,700	114,600
Shaft sealing	90	110	200
Total Field Costs	705,000	600,000	1,305,000
Architect-Engineer Services			56,000
Owner's Costs			156,000
Contingency			389,000
TOTAL FACILITY COST			1,906,000

TABLE 7.B.25. Expenditure Schedule for Continuous Mining - Salt Repository - U-Only Recycle; Pu to HLW

Time, years	Material Percent Expended	Material Cumulative	Labor Percent Expended	Labor Cumulative
1	--	--	--	--
2	1.4	1.4	--	--
3	8.7	10.1	3.0	3.0
4	8.7	18.8	6.4	9.4
5	5.7	24.5	6.4	15.8
6	5.7	30.2	6.4	22.2
7	5.7	35.9	6.4	28.6
8	5.7	41.6	6.4	35.0
9	5.7	47.3	6.4	41.4
10	5.7	53.0	6.4	47.8
11	5.7	58.7	6.4	54.2
12	5.7	64.4	6.4	60.6
13	5.7	70.1	6.4	67.0
14	5.7	75.8	6.4	73.4
15	5.7	81.5	6.4	79.8
16	5.7	87.2	6.4	86.2
17	5.7	92.9	6.4	92.6
18	5.7	98.6	6.4	99.0
19	1.4	100.0	1.0	100.0

TABLE 7.B.26. Repository Expenditure Schedule for Continuous Mining - Salt Repository - U-Only Recycle; Pu Stored

Time, years	Material Percent Expended	Material Cumulative	Labor Percent Expended	Labor Cumulative
1	--	--	--	--
2	0.8	0.8	--	--
3	6.3	7.1	2.3	2.3
4	6.3	13.4	4.6	6.9
5	4.3	17.7	4.6	11.5
6	4.3	22.0	4.6	16.1
7	4.3	26.3	4.6	20.7
8	4.3	30.6	4.6	25.3
9	4.3	34.9	4.6	29.9
10	4.3	39.2	4.6	34.5
11	4.3	43.5	4.6	39.1
12	4.3	47.8	4.6	43.7
13	4.3	52.1	4.6	48.3
14	4.3	56.4	4.6	52.9
15	4.3	60.7	4.6	57.5
16	4.3	65.0	4.6	62.1
17	4.3	69.3	4.6	66.7
18	4.3	73.6	4.6	71.3
19	4.3	77.9	4.6	75.9
20	4.3	82.2	4.6	80.5
21	4.3	86.5	4.6	85.1
22	4.3	90.8	4.6	89.7
23	4.3	95.1	4.6	94.3
24	4.3	99.4	4.6	98.9
25	0.6	100.0	1.1	100.0

TABLE 7.B.27. Repository Expenditure Schedule for Continuous Mining - Salt Repository - U and Pu Recycle

Time, years	Material Percent Expended	Material Cumulative	Labor Percent Expended	Labor Cumulative
1	--	--	--	--
2	1.7	1.7	--	--
3	6.9	8.6	2.0	2.0
4	6.9	15.5	5.1	7.1
5	4.6	20.1	5.1	12.2
6	4.6	24.7	5.1	17.3
7	4.6	29.3	5.1	22.4
8	4.6	33.9	5.1	27.5
9	4.6	38.5	5.1	32.6
10	4.6	43.1	5.1	37.7
11	4.6	47.7	5.1	42.8
12	4.6	52.3	5.1	47.9
13	4.6	56.9	5.1	53.0
14	4.6	61.5	5.1	58.1
15	4.6	66.1	5.1	63.2
16	4.6	70.7	5.1	68.3
17	4.6	75.3	5.1	73.4
18	4.6	79.9	5.1	78.5
19	4.6	84.5	5.1	83.6
20	4.6	89.1	5.1	88.7
21	4.6	93.7	5.1	93.8
22	4.6	98.3	5.1	98.9
23	1.7	100.0	1.1	100.0

TABLE 7.B.28. Expenditure Schedule for Continuous Mining - Granite Repository - All Recycle Cases

Time, years	Material Percent Expended	Material Cumulative	Labor Percent Expended	Labor Cumulative
1	--	--	--	--
2	1.6	1.6	--	--
3	6.6	8.2	2.4	2.4
4	6.6	14.8	4.8	7.2
5	4.4	19.2	4.8	12.0
6	4.4	23.6	4.8	16.8
7	4.4	28.0	4.8	21.6
8	4.4	32.4	4.8	26.4
9	4.4	36.8	4.8	31.2
10	4.4	41.2	4.8	36.0
11	4.4	45.6	4.8	40.8
12	4.4	50.0	4.8	45.6
13	4.4	54.4	4.8	50.4
14	4.4	58.8	4.8	55.2
15	4.4	63.2	4.8	60.0
16	4.4	67.6	4.8	64.8
17	4.4	72.0	4.8	69.6
18	4.4	76.4	4.8	74.4
19	4.4	80.8	4.8	79.2
20	4.4	85.2	4.8	84.0
21	4.4	89.6	4.8	88.8
22	4.4	94.0	4.8	93.6
23	4.4	98.4	4.8	98.4
24	1.6	100.0	1.6	100.0

TABLE 7.B.29. Expenditure Schedule for Continuous Mining - Shale Repository - All Recycle Cases

Time, years	Material: Percent Expended	Material: Cumulative	Labor: Percent Expended	Labor: Cumulative
1	--	--	--	--
2	1.6	1.6	--	--
3	9.7	11.3	3.7	3.7
4	9.7	21.0	7.3	11.0
5	6.5	27.5	7.3	18.3
6	6.5	34.0	7.3	25.6
7	6.5	40.5	7.3	32.9
8	6.5	47.0	7.3	40.2
9	6.5	53.5	7.3	47.5
10	6.5	60.0	7.3	54.8
11	6.5	66.5	7.3	62.1
12	6.5	73.0	7.3	69.4
13	6.5	79.5	7.3	76.7
14	6.5	86.0	7.3	84.0
15	6.5	92.5	7.3	91.3
16	6.5	99.0	7.3	98.6
17	1.0	100.0	1.4	100.0

TABLE 7.B.30. Expenditure Schedule for Continuous Mining - Basalt Repository - All Recycle Cases

Time, years	Material: Percent Expended	Material: Cumulative	Labor: Percent Expended	Labor: Cumulative
1	--	--	--	--
2	1.1	1.1	--	--
3	7.3	8.4	2.7	2.7
4	7.3	15.7	5.3	8.0
5	4.9	20.6	5.3	13.3
6	4.9	25.5	5.3	18.6
7	4.9	30.4	5.3	23.9
8	4.9	35.3	5.3	29.2
9	4.9	40.2	5.3	34.5
10	4.9	45.1	5.3	39.8
11	4.9	50.0	5.3	45.1
12	4.9	54.9	5.3	50.4
13	4.9	59.8	5.3	55.7
14	4.9	64.7	5.3	61.0
15	4.9	69.6	5.3	66.3
16	4.9	74.5	5.3	71.6
17	4.9	79.4	5.3	76.9
18	4.9	84.3	5.3	82.2
19	4.9	89.2	5.3	87.5
20	4.9	94.1	5.3	92.8
21	4.9	99.0	5.3	98.1
22	1.0	100.0	1.9	100.0

APPENDIX 7C

CONSTRUCTION AND MINING COST ALLOCATIONS FOR GEOLOGIC REPOSITORIES FOR FOUR FUEL CYCLES

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CONSTRUCTION AND MINING COST ALLOCATIONS FOR GEOLOGIC REPOSITORIES FOR FOUR FUEL CYCLES

The following tables present the construction and mining costs previously given in Appendices 7A and 7B allocated by waste category. These cost allocations do not include the owners' costs which were allocated in the same ratio as mining and construction cost by waste type to total mining and construction cost. Allocations are given for both accelerated and continuous mining alternatives for four fuel cycles: the Once-through cycle; U-Only Recycle - Pu in HLW; U-Only Recycle - Pu Stored; and U and Pu Recycle.

TABLE 7.C.1. Construction and Mining Cost Allocations - Geologic Isolation in Salt Accelerated Mining, $1,000

Facility	Once-Through Cycle	Uranium-Only Recycle Pu to HLW	Uranium-Only Recycle Pu Stored	U-Pu Recycle
Surface Facilities				
Buildings				
- Canister Waste	26,060	39,100	46,660	46,770
- LLW	--	3,420	3,400	3,400
- Non-allocable	44,570	45,040	44,770	44,880
Subtotal	70,630	87,560	94,830	95,050
Major Equipment				
- Canister Waste	29,690	32,040	34,150	34,220
- LLW	--	2,560	2,560	2,560
- Non-allocable	35,110	35,380	35,340	35,430
Subtotal	64,800	69,980	72,050	72,210
Bulk Material				
- Canister Waste	13,550	13,700	13,610	13,650
- LLW	--	--	--	--
- Non-allocable	48,650	52,500	52,190	52,300
Subtotal	62,200	66,200	65,800	65,950
Site Development				
- Canister Waste	1,160	1,180	1,170	1,190
- LLW	--	170	170	170
- Non-allocable	15,670	15,670	15,570	15,600
Subtotal	16,830	17,020	16,910	16,960
TOTAL SURFACE FACILITIES	214,460	240,760	249,590	250,170
Shafts and Hoists				
Major Equipment				
- Canister Waste Shaft	7,650	7,730	7,690	7,700
- LLW Shaft	--	660	660	660
- Men and Material Shaft	15,590	15,760	15,660	15,710
Subtotal	23,240	24,150	24,010	24,070
Shafts and Linings				
- Canister Waste Shaft	30,110	30,440	30,250	30,320
- LLW Shaft	--	6,230	6,190	6,200
- Men and Material Shaft	38,540	38,900	38,670	38,770
- Ventilation Shaft	23,820	24,070	23,930	23,980
Subtotal	92,470	99,640	99,040	99,270
TOTAL SHAFTS AND HOISTS	115,710	123,790	123,050	123,340
Underground Facilities				
Structures				
- HLW/PWR	5,770	1,940	1,930	1,930
- CW, ILW/BWR	3,840	6,800	6,760	6,780
- LLW	--	970	970	970
Subtotal	9,610	9,710	9,660	9,680
Major Equipment				
- HLW/PWR	2,780	930	930	930
- CW, ILW/BWR	1,860	3,520	3,500	3,500
- LLW	--	240	230	240
Subtotal	4,640	4,690	4,660	4,670
Bulk Material				
- HLW/PWR	2,810	2,850	2,830	2,840
- CW, ILW/BWR	1,880	1,430	1,420	1,420
- LLW	--	470	470	470
Subtotal	4,690	4,750	4,720	4,730
TOTAL UNDERGROUND FACILITIES	18,940	19,150	19,040	19,080
Mining				
Major Equipment				
- HLW/PWR	40,970	41,410	41,170	41,250
- CW, ILW/BWR	27,310	20,710	20,590	20,630
- LLW	--	6,900	6,860	6,880
Subtotal	68,280	69,020	68,610	68,760
Mine Construction				
- HLW/PWR	180,670	184,080	208,990	214,970
- CW, ILW/BWR	120,440	92,040	104,490	107,480
- LLW	--	30,680	34,830	35,830
Subtotal	301,110	306,800	348,310	358,280
TOTAL MINING	369,390	375,820	416,920	427,040
Backfilling				
- HLW/PWR	56,700	57,290	66,840	65,620
- CW, ILW/BWR	37,800	28,640	33,420	32,810
- LLW	--	9,550	11,140	10,940
TOTAL BACKFILLING	94,500	95,480	111,400	109,370

TABLE 7.C.2. Construction and Mining Cost Allocations - Geologic Isolation in Salt, Continuous Mining, $1,000

Facility	Once-Through Cycle	Uranium-Only Recycle Pu to HLW	Uranium-Only Recycle Pu Stored	U-Pu Recycle
Surface Facilities				
Buildings				
- Canister Waste	26,240	38,940	47,100	46,870
- LLW	--	3,400	3,430	3,410
- Non-allocable	44,870	44,870	45,180	44,990
Subtotal	71,110	87,210	95,710	95,270
Major Equipment				
- Canister Waste	29,970	31,880	34,520	34,310
- LLW	--	2,550	2,580	2,570
- Non-allocable	33,770	33,760	34,100	33,990
Subtotal	63,740	68,190	71,200	70,870
Bulk Material				
- Canister Waste	13,640	13,640	13,740	13,680
- LLW	--	--	--	--
- Non-allocable	48,070	51,180	51,550	51,300
Subtotal	61,710	64,820	65,290	64,980
Site Development				
- Canister Waste	1,170	1,170	1,180	1,180
- LLW	170	170	170	170
- Non-allocable	15,430	15,430	15,540	15,460
Subtotal	16,770	16,770	16,890	16,810
TOTAL SURFACE FACILITIES	213,330	236,990	249,090	247,930
Shafts and Hoists				
Major Equipment				
- Canister Waste Shaft	7,700	7,700	7,760	7,720
- LLW Shaft	--	660	670	670
- Men and Material Shaft	11,700	11,690	11,780	11,720
Subtotal	19,400	20,050	20,210	20,110
Shafts and Linings				
- Canister Waste Shaft	30,320	30,320	30,540	30,390
- LLW Shaft	--	6,200	6,250	6,220
- Men and Material Shaft	34,680	34,600	34,860	34,690
- Ventilation Shaft	23,980	23,980	24,150	24,040
Subtotal	88,980	95,100	95,800	95,340
TOTAL SHAFTS AND HOISTS	108,380	115,150	116,010	115,450
Underground Facilities				
Structures				
- HLW/PWR	3,870	1,940	1,950	1,940
- CW, ILW/BWR	5,810	6,780	6,830	6,790
- LLW	--	960	970	970
Subtotal	9,680	9,680	9,750	9,700
Major Equipment				
- HLW/PWR	1,870	940	940	940
- CW, ILW/BWR	2,800	3,500	3,530	3,510
- LLW	--	230	230	230
Subtotal	4,670	4,670	4,700	4,680
Bulk Material				
- HLW/PWR	2,840	2,340	2,850	2,840
- CW, ILW/BWR	1,890	1,420	1,430	1,420
- LLW	--	470	480	480
Subtotal	4,730	4,730	4,760	4,740
TOTAL UNDERGROUND FACILITIES	19,080	19,080	19,210	19,120
Mining				
Major Equipment				
- HLW/PWR	41,250	41,250	41,550	41,350
- CW, ILW/BWR	27,500	20,630	20,780	20,670
- LLW	--	6,870	6,920	6,890
Subtotal	68,750	68,750	69,250	68,910
Mine Construction				
- HLW/PWR	181,970	183,470	211,050	215,580
- CW, ILW/BWR	121,310	91,730	105,530	107,790
- LLW	--	30,580	35,170	35,930
Subtotal	303,280	305,780	351,750	359,300
TOTAL MINING	372,030	374,530	421,000	428,210
Backfilling				
- HLW/PWR	40,310	40,350	29,810	30,770
- CW, ILW/BWR	26,870	20,180	14,910	15,390
- LLW	--	6,720	4,970	5,130
TOTAL BACKFILLING	67,180	67,250	49,690	51,290

TABLE 7.C.3. Construction and Mining Cost Allocations - Geologic Isolation in Granite - Accelerated Mining, $1,000

Facility	Once-Through Cycle	Uranium-Only Recycle		U-Pu Recycle
		Pu to HLW	Pu Stored	
Surface Facilities				
Buildings				
- Canister Waste	40,710	50,260	50,260	50,260
- LLW	--	3,330	3,330	3,330
- *Non-allocable*	43,490	43,810	43,810	43,810
Subtotal	84,200	97,400	97,400	97,400
Major Equipment				
- *Canister Waste*	33,680	34,900	34,900	34,900
- LLW	--	2,540	2,540	2,540
- Non-allocable	34,320	34,560	34,560	34,560
Subtotal	68,000	72,000	72,000	72,000
Bulk Material				
- Canister Waste	13,220	13,320	13,320	13,300
- LLW	--	2,400	2,400	2,390
- *Non-allocable*	47,480	48,680	48,680	48,610
Subtotal	60,700	64,400	64,400	64,300
Site Development				
- *Canister Waste*	1,150	1,160	1,160	1,150
- LLW	--	170	160	160
- Non-allocable	15,250	15,270	15,280	15,190
Subtotal	16,400	16,600	16,600	16,500
TOTAL SURFACE FACILITIES	229,300	250,400	250,400	250,200
Shafts and Hoists				
Major Equipment				
- Canister Waste Shaft	9,770	9,820	9,820	9,800
- LLW Shaft	--	1,880	1,880	1,880
- *Men and Material Shaft*	28,630	28,800	28,800	28,720
Subtotal	38,400	40,500	40,500	40,400
Shafts and Linings				
- *Canister Waste Shaft*	13,900	14,000	14,000	13,980
- LLW Shaft	--	7,680	7,680	7,660
- Men and Material Shaft	75,000	75,500	75,550	75,420
- Ventilation Shaft	16,800	16,970	16,970	16,940
Subtotal	105,700	114,200	114,200	114,000
TOTAL SHAFTS AND HOISTS	144,100	154,700	154,700	154,400
Underground Facilities				
Structures				
- HLW/PWR	3,760	1,880	1,880	1,880
- *CW, ILW/BWR*	5,640	6,580	6,580	6,580
- LLW	--	940	940	940
Subtotal	9,400	9,400	9,400	9,400
Major Equipment				
- HLW/PWR	1,800	920	920	920
- CW, ILW/BWR	2,700	3,450	3,450	3,450
- LLW	--	230	230	230
Subtotal	4,500	4,600	4,600	4,600
Bulk Material				
- HLW/PWR	2,760	2,760	2,760	2,760
- *CW, ILW/BWR*	1,840	1,380	1,380	1,380
- LLW	--	460	460	460
Subtotal	4,600	4,600	4,600	4,600
TOTAL UNDERGROUND FACILITIES	18,500	18,600	18,600	18,600
Mining				
Major Equipment				
- HLW/PWR	51,900	52,260	52,260	52,200
- CW, ILW/BWR	34,600	26,130	26,130	26,100
- LLW	--	8,710	8,710	8,700
Subtotal	86,500	87,100	87,100	87,000
Mine Construction				
- HLW/PWR	801,700	541,320	546,200	554,580
- *CW, ILW/BWR*	534,400	270,660	273,100	277,290
- LLW	--	90,220	91,000	92,430
Subtotal	1,336,100	902,200	910,300	924,300
TOTAL MINING	1,422,600	989,300	997,400	1,011,300
Backfilling				
- *HLW/PWR*	146,100	98,940	98,940	100,500
- CW, ILW/BWR	97,400	49,470	49,470	50,250
- LLW	--	16,490	16,490	16,750
TOTAL BACKFILLING	243,500	164,900	164,900	167,500

TABLE 7.C.4. Construction and Mining Cost Allocations - Geologic Isolation in Granite, Continuous Mining, $1,000

Facility	Once-Through Cycle	Uranium-Only Recycle Pu to HLW	Uranium-Only Recycle Pu Stored	U-Pu Recycle
Surface Facilities				
Buildings				
- Canister Waste	40,640	50,220	50,180	50.190
- LLW	--	3,320	3,320	3,330
- Non-allocable	43,360	43,760	43,700	43,780
Subtotal	84,000	97,300	97,200	97,300
Major Equipment				
- Canister Waste	33,570	34,870	34,870	34,860
- LLW	--	2,540	2,540	2,540
- Non-allocable	32,830	33,090	33,090	33,100
Subtotal	66,400	70,500	70,500	70,500
Bulk Material				
- Canister Waste	13,200	13,310	13,310	13,300
- LLW	--	2,390	2,390	2,390
- Non-allocable	46,500	47,600	47,600	47,410
Subtotal	59,700	63,300	63,300	63,100
Site Development				
- Canister Waste	1,130	1,150	1,150	1,150
- LLW	--	160	160	160
- Non-allocable	15,070	15,090	15,090	15,090
Subtotal	16,200	16,400	16,400	16,400
TOTAL SURFACE FACILITIES	226,300	247,500	247,400	247,300
Shafts and Hoists				
Major Equipment				
- Canister Waste Shaft	9,730	9,820	9,810	9,800
- LLW Shaft	--	1,880	1,890	1,880
- Men and Material Shaft	28,570	28,800	28,800	28,720
Subtotal	38,300	40,500	40,500	40,400
Shafts and Linings				
- Canister Waste Shaft	13,870	13,990	13,990	13,960
- LLW Shaft	--	7,670	7,670	7,650
- Men and Material Shaft	55,230	55,690	55,690	55,580
- Ventilation Shaft	16,800	16,950	16,950	16,910
Subtotal	85,900	94,300	94,300	94,100
TOTAL SHAFTS AND HOISTS	124,200	134,800	134,800	134,500
Underground Facilities				
Structures				
- HLW/PWR	3,720	1,880	1,880	1,880
- CW, ILW/BWR	5,580	6,580	6,580	6,580
- LLW	--	940	940	940
Subtotal	9,300	9,400	9,400	9,400
Major Equipment				
- HLW/PWR	1,800	900	900	900
- CW, ILW/BWR	2,700	3,380	3,380	3,380
- LLW	--	220	220	220
Subtotal	4,500	4,500	4,500	4,500
Bulk Material				
- HLW/PWR	2,760	2,760	2,760	2,760
- CW, ILW/BWR	1,840	1,380	1,380	1,380
- LLW	--	460	460	460
Subtotal	4,600	4,600	4,600	4,600
TOTAL UNDERGROUND FACILITIES	18,400	18,500	18,500	18,500
Mining				
Major Equipment				
- HLW/PWR	51,780	52,200	52,200	52,100
- CW, ILW/BWR	34,520	26,100	26,100	26,000
- LLW	--	8,700	8,700	8,700
Subtotal	86,300	87,000	87,000	86,800
Mine Construction				
- HLW/PWR	800,380	540,680	545,520	554,140
- CW, ILW/BWR	533,520	270,290	272,760	276,920
- LLW	--	90,130	90,920	92,340
Subtotal	1,333,900	901,100	909,200	923,400
TOTAL MINING	1,420,200	988,100	996,200	1,010,200
Backfilling				
- HLW/PWR	116,940	79,260	79,260	80,700
- CW, ILW/BWR	77,960	39,630	39,630	40,350
- LLW	--	13,210	13,210	13,450
TOTAL BACKFILLING	194,900	132,100	132,100	134,500

TABLE 7.C.5. Construction and Mining Cost Allocations - Geologic Isolation in Shale - Accelerated Mining, $1,000

Facility	Once-Through Cycle	Uranium-Only Recycle Pu to HLW	Uranium-Only Recycle Pu Stored	U-Pu Recycle
Surface Facilities				
Buildings				
- Canister Waste	30,980	41,780	42,280	42,280
- LLW	--	3,380	3,380	3,380
- Non-allocable	44,220	44,540	44,540	44,540
Subtotal	75,200	89,700	90,200	90,200
Major Equipment				
- Canister Waste	30,930	32,240	32,540	32,540
- LLW	--	2,540	2,540	2,540
- Non-allocable	34,870	35,220	35,220	35,220
Subtotal	65,800	70,000	70,300	70,300
Bulk Material				
- Canister Waste	13,420	13,550	13,550	13,550
- LLW	--	2,440	2,440	2,440
- Non-allocable	48,180	49,510	49,510	49,510
Subtotal	61,600	65,500	65,500	65,500
Site Development				
- Canister Waste	1,170	1,170	1,170	1,170
- LLW	--	170	170	170
- Non-allocable	15,530	15,460	15,460	15,460
Subtotal	16,700	16,800	16,800	16,800
TOTAL SURFACE FACILITIES	219,300	242,000	242,800	242,800
Shafts and Hoists				
Major Equipment				
- Canister Waste Shaft	10,550	10,660	10,660	10,660
- LLW Shaft	--	2,740	2,740	2,740
- Men and Material Shaft	14,750	14,900	14,900	14,900
Subtotal	25,300	28,300	28,300	28,300
Shafts and Linings				
- Canister Waste Shaft	12,340	12,470	12,450	12,450
- LLW Shaft	--	7,600	7,590	7,590
- Men and Material Shaft	28,990	29,300	29,250	29,250
- Ventilation Shaft	14,870	15,030	15,010	15,010
Subtotal	56,200	64,400	64,300	64,300
TOTAL SHAFTS AND HOISTS	81,500	92,700	92,600	92,600
Underground Facilities				
Structures				
- HLW/PWR	3,800	1,930	1,920	1,920
- CW, ILW/BWR	5,700	6,710	6,720	6,720
- LLW	--	960	960	960
Subtotal	9,500	9,600	9,600	9,600
Major Equipment				
- HLW/PWR	1,850	1,000	920	920
- CW, ILW/BWR	2,750	3,500	3,450	3,450
- LLW	--	200	230	230
Subtotal	4,600	4,700	4,600	4,600
Bulk Material				
- HLW/PWR	2,820	2,820	2,820	2,820
- CW, ILW/BWR	1,880	1,410	1,410	1,410
- LLW	--	470	470	470
Subtotal	4,700	4,700	4,700	4,700
TOTAL UNDERGROUND FACILITIES	18,800	19,000	18,900	18,900
Mining				
Major Equipment				
- HLW/PWR	42,060	42,480	42,420	42,420
- CW, ILW/BWR	28,040	21,240	21,210	21,210
- LLW	--	7,080	7,070	7,070
Subtotal	70,100	70,800	70,700	70,700
Mine Construction				
- HLW/PWR	321,640	277,200	281,800	281,800
- CW, ILW/BWR	214,460	138,550	140,900	140,900
- LLW	--	46,350	46,900	46,900
Subtotal	536,100	462,100	469,600	469,600
TOTAL MINING	606,200	532,900	540,300	540,300
Backfilling				
- HLW/PWR	68,520	58,440	58,440	58,440
- CW, ILW/BWR	45,680	29,220	29,220	29,220
- LLW	--	9,740	9,740	9,740
TOTAL BACKFILLING	114,200	97,400	97,400	97,400

TABLE 7.C.6. Construction and Mining Cost Allocations - Geologic Isolation in Shale, Continuous Mining, $1,000

Facility	Once-Through Cycle	Uranium-Only Recycle Pu to HLW	Uranium-Only Recycle Pu Stored	U-Pu Recycle
Surface Facilities				
Buildings				
- Canister Waste	30,940	41,560	42,030	42,030
- LLW	--	3,360	3,360	3,360
- Non-allocable	44,160	44,280	44,310	44,310
Subtotal	75,100	89,200	89,700	89,700
Major Equipment				
- Canister Waste	30,850	32,010	32,370	32,370
- LLW	--	2,530	2,530	2,530
- Non-allocable	33,350	33,560	33,600	33,600
Subtotal	64,200	68,100	68,500	68,500
Bulk Material				
- Canister Waste	13,400	13,470	13,490	13,490
- LLW	--	2,420	2,430	2,430
- Non-allocable	47,200	48,110	48,180	48,180
Subtotal	60,600	64,000	64,100	64,100
Site Development				
- Canister Waste	1,150	1,160	1,160	1,160
- LLW	--	170	170	170
- Non-allocable	15,350	15,270	15,270	15,270
Subtotal	16,500	16,600	16,600	16,600
TOTAL SURFACE FACILITIES	216,400	237,900	238,900	238,900
Shafts and Hoists				
Major Equipment				
- Canister Waste Shaft	10,550	10,630	10,630	10,630
- LLW Shaft	--	2,710	2,710	2,710
- Men and Material Shaft	14,750	14,860	14,860	14,860
Subtotal	25,300	28,200	28,200	28,200
Shafts and Linings				
- Canister Waste Shaft	12,250	12,340	12,360	12,360
- LLW Shaft	--	7,520	7,530	7,530
- Men and Material Shaft	22,470	22,620	22,660	22,660
- Ventilation Shaft	14,880	14,920	14,950	14,950
Subtotal	49,600	57,400	57,500	57,500
TOTAL SHAFTS AND HOISTS	74,900	85,600	85,700	85,700
Underground Facilities				
Structures				
- HLW/PWR	3,800	1,900	1,900	1,900
- CW, ILW/BWR	5,700	6,650	6,650	6,650
- LLW	--	950	950	950
Subtotal	9,500	9,500	9,500	9,500
Major Equipment				
- HLW/PWR	1,840	920	920	920
- CW, ILW/BWR	2,760	3,450	3,450	3,450
- LLW	--	230	230	230
Subtotal	4,600	4,600	4,600	4,600
Bulk Material				
- HLW/PWR	2,760	2,820	2,820	2,820
- CW, ILW/BWR	1,840	1,410	1,410	1,410
- LLW	--	470	470	470
Subtotal	4,600	4,700	4,700	4,700
TOTAL UNDERGROUND FACILITIES	18,700	18,800	18,800	18,800
Mining				
Major Equipment				
- HLW/PWR	42,000	42,240	42,240	42,240
- CW, ILW/BWR	28,000	21,120	21,120	21,120
- LLW	--	7,040	7,040	7,040
Subtotal	70,000	70,400	70,400	70,400
Mine Construction				
- HLW/PWR	320,800	275,580	280,420	280,420
- CW, ILW/BWR	213,900	137,740	140,210	140,210
- LLW	--	46,080	46,670	46,670
Subtotal	534,700	459,400	467,300	467,300
TOTAL MINING	604,700	529,800	537,700	537,700
Backfilling				
- HLW/PWR	55,380	46,740	46,740	46,740
- CW, ILW/BWR	36,920	23,370	23,370	23,370
- LLW	--	7,790	7,790	7,790
TOTAL BACKFILLING	92,300	77,900	77,900	77,900

TABLE 7.C.7. Construction and Mining Cost Allocations - Geologic Isolation in Basalt - Accelerated Mining, $1,000

Facility	Once-Through Cycle	Uranium-Only Recycle Pu to HLW	Uranium-Only Recycle Pu Stored	U-Pu Recycle
Surface Facilities				
Buildings				
- Canister Waste	40,570	49,960	49,960	49,960
- LLW	--	3,340	3,330	3,330
- Non-allocable	43,330	43,600	43,610	43,610
Subtotal	83,900	96,900	96,900	96,900
Major Equipment				
- Canister Waste	33,580	34,700	34,700	34,700
- LLW	--	2,540	2,540	2,540
- Non-allocable	34,220	34,360	34,360	34,360
Subtotal	67,800	71,600	71,600	71,600
Bulk Material				
- Canister Waste	13,190	13,270	13,340	13,290
- LLW	--	2,390	2,400	2,390
- Non-allocable	47,310	48,540	48,760	48,520
Subtotal	60,500	64,200	64,500	64,200
Site Development				
- Canister Waste	1,150	1,150	1,160	1,160
- LLW	--	160	170	170
- Non-allocable	15,250	15,190	15,270	15,170
Subtotal	16,400	16,500	16,600	16,500
TOTAL SURFACE FACILITIES	228,600	249,200	249,600	249,200
Shafts and Hoists				
Major Equipment				
- Canister Waste Shaft	11,050	11,130	11,190	11,140
- LLW Shaft	--	2,150	2,160	2,160
- Men and Material Shaft	32,650	32,820	33,050	32,910
Subtotal	43,700	46,100	46,400	46,200
Shafts and Linings				
- Canister Waste Shaft	22,640	22,800	22,910	22,820
- LLW Shaft	--	8,720	8,760	8,730
- Men and Material Shaft	77,910	78,400	78,860	78,550
- Ventilation Shaft	18,650	18,780	18,870	18,800
Subtotal	119,200	128,700	129,400	128,900
TOTAL SHAFTS AND HOISTS	162,900	174,800	175,800	175,100
Underground Facilities				
Structures				
- HLW/PWR	3,800	1,880	1,880	1,880
- CW, ILW/BWR	5,600	6,580	6,580	6,580
- LLW	--	940	940	940
Subtotal	9,400	9,400	9,400	9,400
Major Equipment				
- HLW/PWR	1,800	900	900	900
- CW, ILW/BWR	2,700	3,380	3,380	3,380
- LLW	--	220	220	220
Subtotal	4,500	4,500	4,500	4,500
Bulk Material				
- HLW/PWR	2,760	2,760	2,760	2,760
- CW, ILW/BWR	1,840	1,380	1,360	1,360
- LLW	--	460	460	460
Subtotal	4,600	4,600	4,600	4,600
TOTAL UNDERGROUND FACILITIES	18,500	18,500	18,500	18,500
Mining				
Major Equipment				
- HLW/PWR	51,720	52,080	52,320	52,140
- CW, ILW/BWR	34,480	26,040	26,160	26,070
- LLW	--	8,680	8,720	8,690
Subtotal	86,200	86,800	87,200	86,900
Mine Construction				
- HLW/PWR	977,280	645,780	633,840	650,760
- CW, ILW/BWR	651,520	322,890	316,920	325,380
- LLW	--	107,630	105,640	108,460
Subtotal	1,628,800	1,076,300	1,056,400	1,084,600
TOTAL MINING	1,715,000	1,163,100	1,143,600	1,171,500
Backfilling				
- HLW/PWR	174,600	115,440	112,500	115,620
- CW, ILW/BWR	116,400	57,720	56,250	57,810
- LLW	--	19,240	18,750	19,270
TOTAL BACKFILLING	291,000	192,400	187,500	192,700

TABLE 7.C.8. Construction and Mining Cost Allocations - Geologic Isolation in Basalt, Continuous Mining, $1,000

Facility	Once-Through Cycle	Uranium-Only Recycle Pu to HLW	Uranium-Only Recycle Pu Stored	U-Pu Recycle
Surface Facilities				
Buildings				
- Canister Waste	40,510	49,900	49,850	50,110
- LLW	--	3,340	3,320	3,330
- Non-allocable	43,090	43,460	43,430	43,660
Subtotal	83,600	96,700	96,600	97,100
Major Equipment				
- Canister Waste	33,460	34,630	34,610	34,770
- LLW	--	2,540	2,530	2,540
- Non-allocable	32,740	32,930	32,860	33,090
Subtotal	66,200	70,100	70,000	70,400
Bulk Material				
- Canister Waste	13,100	13,250	13,310	13,330
- LLW	--	2,380	2,390	2,400
- Non-allocable	46,300	47,370	47,600	47,570
Subtotal	59,400	63,000	63,300	63,300
Site Development				
- Canister Waste	1,130	1,140	1,140	1,140
- LLW	--	160	160	160
- Non-allocable	14,970	15,000	15,100	15,200
Subtotal	16,100	16,300	16,300	16,400
TOTAL SURFACE FACILITIES	225,300	246,100	246,200	247,200
Shafts and Hoists				
Major Equipment				
- Canister Waste Shaft	11,000	11,110	11,160	11,180
- LLW Shaft	--	2,140	2,150	2,150
- Men and Material Shaft	32,500	32,750	32,990	32,970
Subtotal	43,500	46,000	46,300	46,300
Shafts and Linings				
- Canister Waste Shaft	15,700	15,800	15,900	15,890
- LLW Shaft	--	8,700	8,700	8,750
- Men and Material Shaft	63,200	63,860	64,200	64,210
- Ventilation Shaft	18,600	18,740	18,800	18,850
Subtotal	97,500	107,100	107,600	107,700
TOTAL SHAFTS AND HOISTS	141,000	153,100	153,900	154,000
Underground Facilities				
Structures				
- HLW/PWR	5,580	1,860	1,880	1,880
- CW, ILW/BWR	3,720	6,510	6,580	6,580
- LLW	--	930	940	940
Subtotal	9,300	9,300	9,400	9,400
Major Equipment				
- HLW/PWR	1,800	900	900	900
- CW, ILW/BWR	2,700	3,380	3,380	3,380
- LLW	--	220	220	220
Subtotal	4,500	4,500	4,500	4,500
Bulk Material				
- HLW/PWR	1,840	2,760	2,760	2,760
- CW, ILW/BWR	2,760	1,380	1,380	1,380
- LLW	--	460	460	460
Subtotal	4,600	4,600	4,600	4,600
TOTAL UNDERGROUND FACILITIES	18,400	18,400	18,500	18,500
Mining				
Major Equipment				
- HLW/PWR	51,480	51,960	52,200	52,300
- CW, ILW/BWR	34,320	25,980	26,100	26,100
- LLW	--	8,660	8,700	8,700
Subtotal	85,800	86,600	87,000	87,100
Mine Construction				
- HLW/PWR	972,960	644,460	632,220	653,160
- CW, ILW/BWR	648,640	322,230	316,110	326,580
- LLW	--	107,410	105,370	108,860
Subtotal	1,621,600	1,074,100	1,053,700	1,088,600
TOTAL MINING	1,707,400	1,160,700	1,140,700	1,175,700
Backfilling				
- HLW/PWR	139,140	92,220	89,820	92,760
- CW, ILW/BWR	92,760	46,110	44,910	46,380
- LLW	--	15,370	14,970	15,460
TOTAL BACKFILLING	231,900	153,700	149,700	154,600

APPENDIX 7D

25-YEAR READY RETRIEVABILITY OF SPENT FUEL

APPENDIX 7.D

25-YEAR READY RETRIEVABILITY OF SPENT FUEL

Extending the spent fuel readily retrievable period to 25 years requires that special steps be taken to ensure accessibility of emplacement rooms and to ensure the integrity of the spent fuel canisters.

The repository design parameter of greatest concern in maintaining room accessibility is near field local thermal density (kW/acre). By decreasing the amount of thermal energy stored in the rooms, thermal stresses in the ceiling and supporting pillars are reduced to the point where room opening stability can be reasonably assured for the 25-year period. Canister integrity is maintained during the retrievability period by using steel sleeves to line emplacement holes and trenches. The sleeves prevent corrosion and protect the canisters from mechanical damage if rock fracture within the holes and trenches should occur. Sleeves may not be required for retrievability in non-salt media; however, until specific sites are identified and thoroughly characterized the need for sleeves will be assumed.

Table 7.D.1 lists near field local thermal densities for 25-year ready retrievability of spent fuel at the conceptual repositories located in salt, granite, shale, and basalt formations. Consistent with the conservative approach taken in the 5-year readily retrievable case, the values in Table 7.D.1 are two thirds of the calculated maximum acceptable thermal densities for 25-year ready retrievability in Reference 1.

TABLE 7.D.1. Near Field Local Thermal Densities[a] for 25-Year Ready Retrievability of Spent Fuel

Near Field Allowable Thermal Loading, kW/acre			
Salt	Granite	Shale	Basalt
24	53	36	53

a. These densities are conservative values that are 2/3 of the calculated densities.

As discussed in Section 7.3, Thermal Criteria, the criteria controlling placement of spent fuel in salt with 5-year ready retrievability is the far field average thermal density. However, in the case of 25-year ready retrievability, near field local thermal density becomes the controlling criteria because maintaining room and pillar stability for 25 years requires a more restrictive thermal density than is needed to limit long term uplift.

The reduced thermal densities listed in Table 7.D.1, require the number of spent fuel canisters emplaced per room (and in the overall repository) be reduced by an amount proportional to the reduction in the near field local thermal densities. This results in approximately a factor of two decrease in the contents of the salt and shale repositories (50 kW/acre ÷ 24 kW/acre for salt and 80 kW/acre ÷ 36 kW/acre for shale) and approximately a factor of 2.5 decrease at the granite and basalt repositories (130 kW/acre ÷ 53 kW/acre for granite and basalt).

An additional concern for the repository in salt is the creep closure of rooms over the 25-year period of ready retrievability. To compensate for this, room ceiling heights are increased to 7.6 m (25 ft) for 25-year ready retrievability (6.7 m (22 ft) for 5-year ready retrievability).

An alternative approach to 25-year retrievability is to provide heat removal from the mine by continuously ventilating emplacement rooms. This technique could allow higher thermal densities by removing heat from the rock formation to keep room and pillar stresses within acceptable limits. Additional details of this approach are provided in Y/OWI/TM-44.[1]

Canister integrity does not appear to be an additional problem for 25-year ready retrievability. Twenty-five-year ready retrievability thicker sleeves may be required than for 5-year ready retrievability.

The unit cost for providing 25-year ready retrievability of emplaced spent fuel elements at a repository located in salt is $78/kgHM (mid-1978 dollars) compared to $46/kgHM for 5-year retrievability. The primary reason for this difference in cost is the reduction of repository waste capacity by about a factor of two for the 25-year retrievability option. Another contribution to the higher cost is $60 million for additional mining and backfilling that is necessary as a result of increased ceiling height for the repository in salt. Use of sleeves for all emplaced wastes also costs an extra $3.4 million annually. Unit costs for 25-year retrievable emplacement of spent fuel in the other rock media would also increase although additional mining to increase ceiling height would not be required.

REFERENCES FOR APPENDIX 7.D

1. Union Carbide Corporation, Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Y/OWI/TM-44, Office of Waste Isolation, Union Carbide Corporation, Nuclear Division Oak Ridge, TN, 1978.

8.0 DECOMMISSIONING OF RETIRED FACILITIES

8.0 DECOMMISSIONING OF RETIRED FACILITIES

Nuclear power plants and postfission fuel cycle facilities become contaminated during power production and fuel cycle and waste treatment operations. Upon retirement, those facilities enter the fuel cycle waste stream. This section reviews possible alternatives for managing retired facilities and procedures for decommissioning.

Decommissioning is defined as the measures taken at the end of the facility's operating life to assure the continued protection of the public from any residual radioactivity or other potential hazards present in the facility. Two basic decommissioning approaches are considered:

- Immediate Dismantlement - Radioactive materials are removed from the facility and the facility is disassembled and decontaminated following final cessation of production operations. After decommissioning, the property is released for unrestricted use.
- Safe Storage with Deferred Dismantlement - Radioactive materials and contaminated areas are secured and structures and equipment are maintained as necessary at the facility to protect the public from the residual radioactivity. Dismantlement is deferred until radioactive decay processes reduce the radioactivity levels within the facility. After dismantlement, the property is released for unrestricted use.

In this section deferred dismantlement is used as a generic term that includes whatever actions necessary at some future time to terminate the facility's nuclear license and release the property for unrestricted use. These actions range from radiation surveys, which show that the residual radioactivity has decayed to reasonable levels, to disassembly and removal of radioactive materials.

Safe storage includes a broad range of possible decommissioning methods. Initial decommissioning activities remove the facility from service and place it in a condition that will protect the public until the facility is dismantled. These initial activities are followed by a continuing care period. Maintenance and surveillance activities during this period assure that the facility remains in this safe condition. Selection of the type of initial activities used to place the facility in safe storage involves economic tradeoffs with the activities required during the interim care period. Minimal removal and fixation of residual radioactivity could be followed by a maintenance and surveillance program using active protection systems (sometimes called layaway) or passive systems (mothballing). Extensive cleanup and decontamination with installation of hardened passive protection systems (temporary entombment) could be followed by a very limited maintenance and surveillance program. The procedures selected for a particular facility depend on the conditions within the facility at shutdown and the period of time that the facility is expected to remain in safe storage prior to dismantlement.

There has been a minimum of experience in decommissioning commercial nuclear facilities. Detailed studies of the procedures, cost, and safety of decommissioning nuclear power plants and fuel cycle facilites are underway at the Pacific Northwest Laboratory under the sponsorship of the Fuel Process Systems Standards Branch, Division of Engineering Standards, U.S. Nuclear Regulatory Commission (NRC). Much of the information presented here is based on preliminary results and insights gathered from these studies. Therefore, the information presented in this section should be considered preliminary in nature, and the results may change when the detailed

studies are completed. However, it is believed that the information presented does provide a reasonable basis for estimating the environmental effects of decommissioning at a reference facility.

This section presents a schedule of events, a manpower estimate, a cost estimate and a description of the waste generated for two alternative decommissioning modes at each of the four reference facilities considered in this study. For some facilities, immediate dismantlement mode may be the only viable decommissioning alternative. For these facilities, an alternative (usually hardened safe storage with an extended interim care period) is included for comparison, even though this alternative would probably not be used. It should also be noted that transportation and disposal costs for decommissioning wastes are being considered elsewhere in this document and are not included in the decommissioning cost estimates. These costs contribute substantially to the total decommissioning costs, particularly for dismantlement.

8.1 BASIC ASSUMPTIONS FOR DECOMMISSIONING

8.1 BASIC ASSUMPTIONS FOR DECOMMISSIONING

The information presented in this section is contingent on a series of basic assumptions. Most of these assumptions are made to insure consistency with other sections of this report, and others serve to simplify the analysis or delineate its limitations. The assumptions that apply to all the facilities considered are outlined below. Others are presented in the discussions on decommissioning the individual facilities in Sections 8.3-8.6.

- The decommissioning plans, costs, and impacts are based on the reference facility descriptions presented in Section 3.2. The basic facility descriptions were expanded to include components representative of a complete operating plant. Additions to the basic facility description that have been made for the purposes of estimating decommissioning costs and impacts are outlined in the discussion on the individual facility.
- The decommissioning plans and cost estimates are based on currently available decommissioning technology. Future improvements in this technology could significantly reduce decommissioning costs.
- The decommissioning plans assume that current regulations regarding release of radioactive material to the environment[1] will remain in effect. Changes in these regulations could affect the plans and cost estimates presented for some of the facilities.
- Current operating practices at commercial waste burial grounds are assumed for the purposes of describing acceptable waste forms for non-TRU wastes. Changes in those practices could affect procedures used.
- The decommissioning cost estimates include allowances for labor, materials, utilities and services, owner overheads and expenses, and onsite treatment and packaging of radioactive wastes. The cost of shipping and disposal of the packaged waste is considered in other sections of this report and is not included in the decommissioning cost estimates.
- Equipment for treatment of non-high level wastes is assumed to be installed at each facility. This equipment includes accommodations for volume reduction of combustible waste and immobilization of wet wastes and treated combustible wastes contaminated with transuranic elements.
- Each facility is assumed to have experienced a typical operating history before shutdown, that is, minor spills of radioactive materials occurred in the plant but no major accidents.
- The decommissioning cost estimates assume that decommissioning proceeds with a minimum of difficulties. A 25% contingency allowance is added to the cost estimates, so that the total costs presented are believed to be reasonable values for performing the operations described. With the contingency added there is an approximately equal chance that the actual decommissioning cost would be greater or less than the cost estimate presented.

- The activities carried out to shut down the facility at the end of its operating lifetime are not included in the decommissioning costs.
- In general, no cost has been included for the disposition of nonradioactive portions of the facilities, such as power substations, warehouses, and office facilities. It is assumed that these facilities can be used further by the facility owner or removed for salvage at no cost to the owner.

8.2 DECOMMISSIONING PROCEDURES

8.2 DECOMMISSIONING PROCEDURES

Decommissioning of retired nuclear facilities generally includes four types of activities: 1) planning and preparation, 2) decommissioning operations and supporting activities, 3) continuing care, and 4) final decommissioning.

The planning and preparation activities are typically carried out during the final one to two years of facility operation. The decommissioning staff is assembled; decommissioning plans and safety analysis and environmental reports are prepared and submitted to the NRC for review; special equipment is designed and fabricated or procured; detailed working procedures and health and safety requirements are developed; and cost estimates are made. A variety of predecommissioning activities are carried out during this phase as part of plant shutdown operations. These activities include removal of bulk quantities of process materials, radioactive materials, and nonessential equipment from the site and renovating effluent control systems that are necessary for decommissioning.

Decommissioning operations and support activities are illustrated in Figure 8.2.1. The chart is generally applicable to both dismantlement and safe storage activities, although the effort and manpower required in each area varies. In addition to direct decommissioning operations, other activities carried out during decommissioning include: development and implementation of the radiation and industrial safety program; maintenance and operation of safety-related facility equipment and other equipment required for the decommissioning; project accounting; management, administration, and procurement and distribution of equipment and supplies; and maintenance of site and facility security.

The continuing care activities required for a facility placed in safe storage are illustrated in Figure 8.2.2. The general types of activities carried out during this period include: operation and maintenance of safety-related equipment retained in operation during the continuing care period, radiation and environmental monitoring programs, security of the decommissioned facility, and inspection of the facility and safety-related equipment and auditing of other continuing care activities to protect the public and environment from the residual radioactivity at the site.

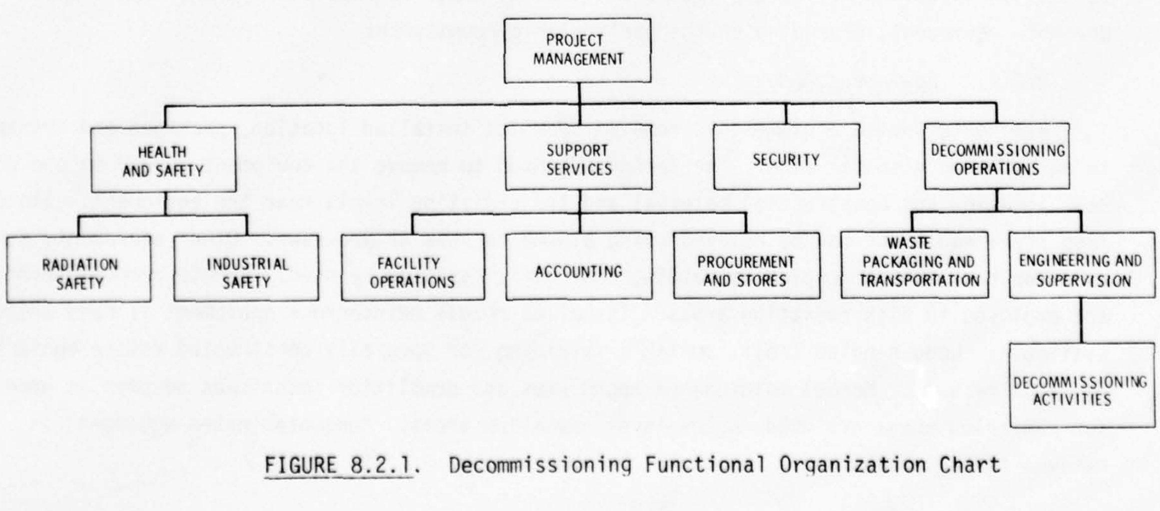

FIGURE 8.2.1. Decommissioning Functional Organization Chart

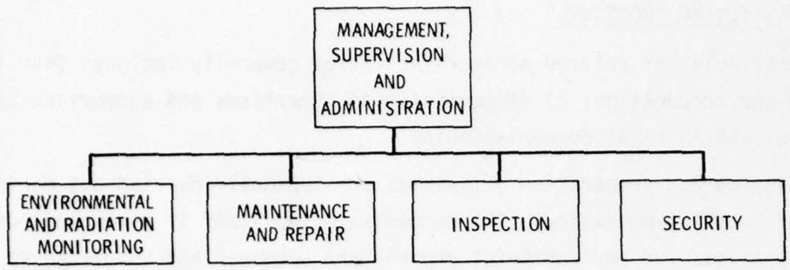

FIGURE 8.2.2. Functional Organization for Interim Care of Retired Facilities

Final decommissioning activities are required at the end of the continuing care period to permit termination of the facility license. These activities may range from performance of radiation surveys to assure that all radioactivity has decayed to nonhazardous levels to removal of all residual radioactivity to an approved disposal site.

The following sections discuss the specific procedures associated with the decommissioning operations portion of the two basic decommissioning modes.

8.2.1 Procedures Associated With Immediate Dismantlement

The activities required to dismantle a retired nuclear facility are discussed below.

8.2.1.1 Chemical Decontamination

Chemical decontamination is carried out to reduce radiation levels, to render residual contamination relatively immobile, and in some facilities, to recover residual amounts of valuable materials. The extent to which chemical decontamination is used at a particular facility depends on a variety of factors including: the radiation dose rates in the facility, the potential effectiveness of chemical decontamination, the amount of chemical decontamination equipment installed in the facility, and the availability of facilities for treating, packaging, and disposing of the chemical decontamination solutions. Decontamination solutions used may range from corrosive acids to detergents to high pressure water or steam. The decontamination solutions may be applied remotely using installed equipment, manually, or with portable equipment, depending on the particular circumstances.

8.2.1.2 Equipment Removal

All contaminated equipment is removed from its installed location, packaged and transported to an approved disposal site. The techniques used to remove the equipment depend on the equipment location and construction material and the radiation levels near the equipment. Stainless steel equipment can be removed using plasma torches or arc-saws. Other equipment, such as power hack saws or explosive cutting equipment, may also be used. Remote removal techniques are employed in high radiation areas. Installed remote maintenance equipment is used where available. Long-handled tools, portable shielding, or specially-constructed remote equipment may also be used. Normal maintenance techniques and demolition techniques adapted as necessary for radiation areas are used in low-level radiation areas. Noncontaminated equipment is removed for salvage.

8.2.1.3 Mechanical Decontamination of Structures

In order to insure that all radioactivity is removed from a facility, it may be necessary to remove portions of the structures. Activated structural materials are entirely removed. Generally, the surface layers are removed from structures with surface contamination. Some noncontaminated materials may have to be removed to insure that all contamination is removed. After removal, contaminated or activated structural materials are packaged and shipped offsite for disposal.

Stainless steel structural components or liners may be removed by sectioning in place with plasma torches, arc saws, or explosives. Contaminated concrete can be removed with explosives,(4) by drilling and rock-splitting,(5) or by jackhammering. Explosives are usually preferred for removal of contamination from large concrete surfaces. Jackhammers or hand tools are generally used for small areas. Rock splitters may be used for moderately sized areas or on large areas where explosives are not desirable. These techniques may be used remotely or by direct contact, depending on radiation levels in the area being decontaminated.

8.2.1.4 Demolition and Site Restoration

Demolition of noncontaminated facility structures is not required from a radiological safety standpoint. Building demolition may be carried out at some sites either because building structural integrity has been degraded by the mechanical decontamination procedures or to permit alternate uses of the site. Generally, conventional demolition techniques such as explosives and impact balls are employed, unless site-specific conditions restrict their use. Concrete rubble may be used as backfill at the site or removed for commercial disposal. Building components are salvaged when it is economically feasible.

Site restoration activities include a final site radiation survey, backfilling of excavations, grading and contouring of the soil, and planting soil-stabilizing vegetation. The extent to which these activities are carried out depends on the anticipated use of the site after decommissioning is completed.

8.2.2 Procedures Associated With Safe Storage

Four general types of activities are carried out to place a retired facility in safe storage: 1) chemical and mechanical decontamination and fixing of residual radioactivity, 2) equipment deactivation, 3) isolation of contaminated areas, 4) final preparation for continuing care. Each of these activities is discussed below. Continuing care activities begin when these activities have been completed and continue until further decommissioning activities are carried out or all radioactivity in the facility decays to required levels.

8.2.2.1 Chemical and Mechanical Decontamination and Fixing Residual Contamination

Chemical decontamination procedures carried out for safe storage are similar to those described previously for dismantlement. Again, the extent to which chemical decontamination is used depends on the conditions at the particular facility when decommissioning begins.

Mechanical decontamination is used to remove significant amounts of radioactive contamination from areas of the facility that will not be isolated during the continuing care period. Contaminated equipment and piping is generally removed using standard maintenance techniques. Contaminated sections of concrete may be removed with hand tools, jackhammers, or by drilling and rocksplitting, depending on the extent and location of the contamination. The contaminated materials to be removed are generally placed in one of the areas that would be isolated during the continuing care period. Some materials, particularly if they are flammable, may be packaged and shipped offsite for disposal.

Some residual amounts of low-level contamination may be left in areas outside the isolated areas. These areas will typically contain amounts of radioactivity that do not contribute significantly to occupational exposure levels in the facility. This contamination would be immobilized by covering it with paint or other protective coatings to prevent the contamination from becoming airborne.

8.2.2.2 Equipment Deactivation

Typically, only essential safety systems such as automatic radiation detection equipment and fire alarms will remain in operation in the facility during the continuing care period. All other equipment is deactivated and placed in a condition that provides maximum safety with minimum maintenance. Whenever possible, equipment is left in a condition that permits salvage at a later date. Deactivation techniques include closing installed valves, installing blank flanges, and disconnecting electrical power and other utilities. A safety audit of all systems is performed to insure that all flammable, corrosive, and other potentially hazardous materials have been removed.

8.2.2.3 Isolation of Contaminated Areas

Isolation procedures vary significantly between passive and hardened safe storage. The activities for each isolation method are discussed below.

- Passive Safe Storage. Portions of the facility containing significant amounts of contamination are isolated from the remainder of the facility by the installation of air-tight barriers. The barriers may be constructed by welding, by bolting and sealing stainless steel plates, or by closing and securing existing barriers. The barriers are constructed so that considerable effort and/or special tools and equipment are needed to open them. HEPA-filtered vents may be installed in isolated areas to allow for changes in temperature and air pressure.

- Hardened Safe Storage. The purpose of the hardened structure is to isolate the radioactivity remaining at the site from the environment until it has decayed to required levels. The isolation structure design goals include: 1) resistance to damage from natural phenomena such as earthquakes, tornadoes, and floods, 2) minimum maintenance during the continuing care period, 3) resistance to incursion of surface and ground water, 4) resistance to intentional penetration, and 5) minimum radiation dose rates at the surface of the structure. A variety of techniques may be used to accomplish these design goals. A substantial portion of the isolation barrier is usually formed by existing structures in the facility. These structures are typically thick, heavily reinforced

concrete walls around portions of the facility that contained large inventories of radioactive material during plant operation. Isolation of the residual radioactivity at the facility usually involves sealing existing penetrations into these areas.

Piping penetrations into the isolation structure are typically sealed by cutting the pipe inside the wall, sealing the end with a pipe plug or welded blank flange, and filling the recess in the concrete wall with concrete or grout. Expanding concrete may be used in areas where voids might form in the concrete pour. Large metal surfaces may be sealed with welded steel plates. Portions of the facility containing many penetrations into the isolation structure, such as pipe or instrument vaults, may be completely filled with concrete rather than sealing individual penetrations. Reinforced concrete is used where structural strength is required. An epoxy or other suitable adhesive may be used to assure that new concrete pours adhere to existing concrete surfaces. External surfaces of the isolation structure that could be exposed to moisture which might degrade the concrete or steel surfaces are coated with a sealant such as epoxy-coal tar, bitumen or asphalt. The interior volume of the isolation structure is usually dried before the final seal is made. It may also be filled with a dry matrix material such as sand.

8.2.2.4 Final Preparations for the Continuing Care Period

There are important differences between final preparations for continuing care for passive and for hardened safe storage. The activities for each method are discussed below.

- Passive Safe Storage. Most exterior doors into the facility are disabled, typically by welding. High security locks are installed on remaining doors and electronic intruder alarms may be installed to detect unauthorized entry into the facility during the continuing care period. Automatic radiation detection equipment and fire detection and automatic firefighting equipment are inspected and upgraded as necessary. Remote readout equipment are installed for safety systems that would be in operation during the continuing care period. Final radiation surveys are performed. Portions of the facility site that do not contain radioactivity may be prepared for release for other purposes. The facility and site boundary fence may be relocated, although a security fence would be maintained around the facility during the continuing care period.

  After these activities are completed, the continuing care activities will take place until further decommissioning activities are carried out or until all radioactivity in the facility has decayed to levels required by regulations.

- Hardened Safe Storage. Any radioactive or other hazardous materials remaining on the site outside the isolation structure are packaged and transported offsite. This may include, for example, some ventilation equipment that provided contamination control while the isolation structure was being completed. Final radiation surveys of the facility and site are performed. Since major portions of the site and facility may be used for other purposes during the continuing care period, the site boundary and facility security fences would likely be removed or relocated. Warning markers are placed on or near the isolation structure and time capsules containing details of the facility

description, operating history, and decommissioning operations placed at strategic locations. Similar sets of records are left with government agencies. Some nonradioactive portions of the facility may be removed from the site using normal demolition techniques that do not threaten the integrity of the isolation structure. Electrical power, except possibly for some essential lighting circuits, and other utilities are removed. The facility's final condition requires no routine maintenance, with relatively infrequent inspections and radiation and environmental surveys.

After these activities are completed, the minimal continuing care procedures required for facility in hardened safe storage would be initiated. The continuing care period lasts until the radioactivity in the facility has decayed to levels that permit termination of the facility license or until further decommissioning activities are carried out.

Alternative decommissioning modes and the procedures to execute them vary with the facility to which they are applied. Reference facilities considered here are the nuclear power plant, the independent spent fuel storage basin, the fuel reprocessing plant, and the mixed oxide fuel fabrication plant. Basic assumptions are discussed in Section 8.1. Those assumptions that vary with individual facilities are discussed in relation to the particular facility in the following sections.

8.3 DECOMMISSIONING OF A NUCLEAR POWER PLANT

8.3 DECOMMISSIONING OF A NUCLEAR POWER PLANT

Two modes are considered for decommissioning the reference light water reactor (LWR) nuclear power plant: 1) immediate dismantlement, and 2) passive safe storage with deferred dismantlement. For the case of safe storage with deferred dismantlement, it is assumed that the facility is maintained in safe storage for at least 50 years following reactor shutdown to allow decay of the ^{60}Co corrosion product activity deposited throughout the plant systems.

8.3.1 Nuclear Power Plant Description

The reference light water reactor, which is assumed to be a pressurized water reactor (PWR), is described in Section 3.2. In addition to the facilities described, the reference power plant is assumed to contain equipment for remote immobilization, packaging, and loadout of wet wastes; and for compaction, packaging, and loadout of combustible wastes.

8.3.2 Immediate Dismantlement of the Reference Nuclear Power Plant

For dismantlement, all radioactive and contaminated materials are removed from the site, the site is restored to a condition that will permit unrestricted use, and all licensed activities are terminated. Decontaminated structures are demolished at the discretion of the plant owner.

8.3.2.1 Decommissioning Plan and Schedule of Events

A description of the general types of activities required to dismantle a retired facility is given in Section 8.2.1. A simplified schedule of events for the immediate dismantlement of the reference power plant is presented in Figure 8.3.1. Approximately six calendar years are

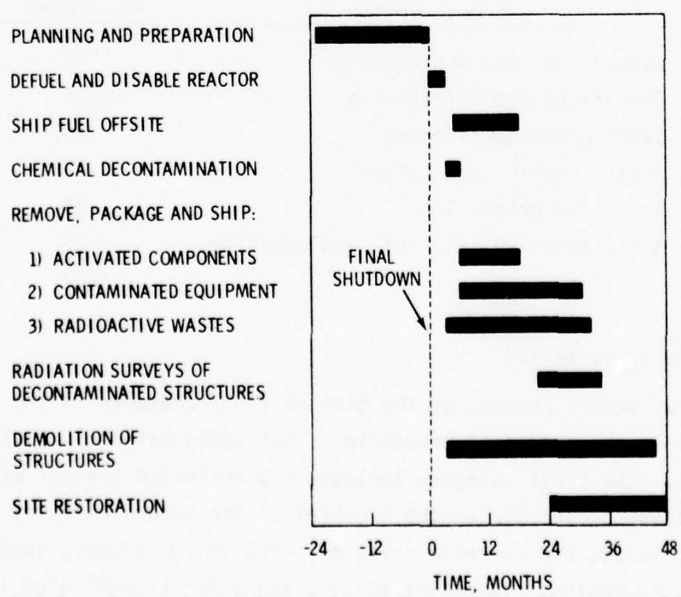

FIGURE 8.3.1. Approximate Schedule of Events for Immediate Dismantlement of the Reference Nuclear Power Plant

required--two years of planning and preparation prior to termination of operations and four years to complete the decontamination and removal of the facility and to restore the site. The post-shutdown period is reduced to less than three years if the facility owner elects not to demolish decontaminated buildings and restore the site.

The general sequence of events begins with a comprehensive radiation survey of the plant, carried out after the reactor is defueled and disabled and the possession-only license is in place. Chemical decontamination of the reactor coolant system (RCS) and the chemical and volume control system (CVCS) is then performed, and removal, packaging, and shipment of radioactive wastes and contaminated and activated components is begun. The work is carried on in parallel in the various structures as much as possible. Each structure is surveyed following dismantlement and decontamination to assure that any residual radioactivity is less than levels allowable for unrestricted use. If demolition is to be performed on the structures, it begins early with the cooling tower and other nonradioactive buildings and continues as buildings are dismantled and decontaminated. Uncontaminated rubble is used for backfilling below-grade cavities.

8.3.2.2 Manpower Requirements

The estimated manpower requirements for the immediate dismantlement of the reference power plant are summarized in Table 8.3.1. These manpower estimates are based on preliminary results of the NRC decommissioning study. This table does not include the personnel needed to accomplish the demolition and site restoration work.

TABLE 8.3.1. Estimated Manpower Requirements for Dismantlement of the Reference Nuclear Power Plant

Job Description	Accumulative Man-Years
Supervisory and engineering	33
Operations and maintenance	29
Skilled decommissioning	76
Health physics and safety	11
Radiation protection	24
Administrative, clerical and security	85
Total	258

8.3.2.3 Radioactive Wastes

The radioactive wastes present at the time of final shutdown in the reference nuclear power plant fall into three general categories: activated material, contaminated material, and process wastes. The first category includes the activated reactor vessel and internal structures and activated concrete in the bioshield. The second category includes contaminated equipment, piping, metal, and concrete surfaces. The third category includes wet and dry filters, ion exchange resins, evaporator bottoms and other contaminated liquids, and miscellaneous combustible wastes such as plastic bags and sheeting, protective clothing, rags,

paper, and wood. Activated and contaminated materials require only sectioning to appropriate sizes for packaging. Process wastes are converted to solidified noncombustible forms whenever possible or compacted in noncombustible containers. Present regulations permit these materials to be sent to shallow land burial facilities. The volumes and curie contents of the radioactive wastes from dismantlement of the reference nuclear power plant are given in Table 8.3.2.

The curies of activity listed in Table 8.3.2 are those present at the time of final reactor shutdown. The principal radionuclides present at reactor shutdown in each category of radioactive waste are given in Table 8.3.3. The waste volumes and radioactivity levels are based on preliminary results from the NRC decommissioning studies.

TABLE 8.3.2. Estimated Waste Volumes from Immediate Dismantlement of the Reference Nuclear Power Plant

Waste Type	Volume, m^3(a)	Waste Description	Radioactivity, Ci/m^3(b)	Waste Disposition
Noncompactable, noncombustible wastes	600	Activated equipment and structural materials	3,500	Packaged, shipped for disposal
	9,900	Contaminated equipment and structural materials	0.06	Packaged, shipped for disposal
Compactable waste and combustible trash	300	Combustible trash	0.01	Treated, packaged, shipped for disposal
	30	Combustible trash	0.01	Packaged, shipped for disposal
	30	HEPA filters	10	Packaged, shipped for disposal
Wet wastes and liquids	50	Spent resins	1,000	Treated, packaged, shipped for disposal
	10	Filter cartridges	500	Treated, packaged, shipped for disposal

a. All waste volumes are for raw waste before treatment or packaging.
b. Table 8.3.3 gives radionuclide content.

8.3.2.4 Routine Radioactive Effluents During Dismantlement

Since ventilation confinement would be maintained on all buildings that contained radioactive materials in the reference nuclear power plant until those materials have been removed and packaged for disposal, no significant releases of airborne radioactivity are anticipated.[6] Aqueous releases would be limited to evaporator condensates and similar liquids that are sampled and shown to contain less radioactivity than is permitted to be released under the facility license.

TABLE 8.3.3. Fractional Abundance of Radionuclide Activities in Radioactive Wastes at Final Reactor Shutdown

Nuclide	Half-Life	Fraction of Total Activity: Activated Material	Fraction of Total Activity: Other Wastes
^{54}Mn	310 d	--	0.042
^{58}Co	72 d	--	0.591
^{60}Co	5.27 yr	0.40	0.082
^{55}Fe	2.7 yr	0.55	--
^{59}Ni	$\sim$80,000 yr	0.003	--
^{63}Ni	$\sim$100 yr	0.05	--
^{51}Cr	28 d	--	0.269
^{59}Fe	45 d	--	0.0.7

8.3.2.5 Cost Estimate

The estimated cost of immediate dismantlement for the reference power plant is presented in Table 8.3.4. These costs were derived using assumptions consistent with the costs estimated developed in a recent fuel reprocessing plant decommissioning study.(7) These estimates were made assuming that the work proceeds with reasonable success and with no major difficulties. A contingency of 25% of total cost has been added to account for unforeseen problems.

TABLE 8.3.4. Estimated Cost of Immediate Dismantlement of the Reference Nuclear Power Plant

Cost Element	Cost, 1000s of mid-1976 Dollars
Dismantlement and decontamination	5,800
Supporting activities	2,100
Utilities	2,800
Equipment and supplies	1,900
Demolition and site restoration	4,700
Nuclear liability insurance	640
Owner costs	4,500
Contingency	4,500
Total	27,000

The total manpower costs, excluding demolition and site restoration, amounted to about $6.2 million of the total given in Table 8.3.4. Supporting activities include such efforts as planning and preparation, chemical decontamination, shipping of spent fuel, radiation protection, quality assurance, and general administrative support. Equipment and supplies include development of special devices and procurement of equipment and expendable supplies. It should be noted that waste transportation and disposal costs are being considered elsewhere in this study and are not included in Table 8.3.4. The total decommissioning costs change significantly when these costs are included.

8.3.2.6 Nonradiological Impacts

In general, the impacts of decommissioning would be much smaller than the impacts of facility construction. These impacts, listed below, are given quantitatively where possible and qualitatively otherwise.

Land Use. The total site would continue to be restricted until all dismantlement and decontamination work is complete and the facility license terminated. From that time forward, unrestricted use of the entire site would be possible. Actual access to the inner exclusion area would be restricted until demolition and site restoration is complete. Approximately 8 ha (20 acres) are included in the inner exclusion area. Land would be required at a commercial burial ground and possibly at a Federal repository to dispose of the radioactive decommissioning wastes.

Water Use. Water usage during decommissioning would be much reduced from operational conditions. Primary use of water would be sanitary systems, flushing, and surface washing during decontamination, and for dust control and ground stabilization during demolition and site restoration.

Material and Equipment. Material and equipment expended during decommissioning would be principally steel, lead, wood, paper and plastic, with the bulk of the material used for shipping containers. Some equipment would be worn out or contaminated beyond cleaning and would be discarded. It is estimated that the following quantities of materials would be expended: steel, $\sim$150 MT; lead, $\sim$400 MT; wood, $\sim$2000 MT; miscellaneous paper and plastic, $\sim$50 MT.

Energy. Electricity would be the major type of energy used during decommissioning, followed by vehicular fuel. An estimated 230,000 MW-hr of electricity would be expended, largely in operating the radwaste evaporators. The vehicular fuel would be used primarily during demolition and site restoration.

Air Quality Effects. No significant effects are anticipated during dismantlement and decontamination. Transitory dusty conditions would be expected during demolition and site restoration, together with vehicular exhaust gases.

Noise Impacts. No significant noise impacts offsite are expected during dismantlement and decontamination. Onsite noise levels are likely to be smaller than during facility operation. Offsite noise levels may be increased during demolition and site restoration and would be comparable to levels encountered during construction.

Effects on Nearby Communities. The principal impacts on the local communities would be reduced payroll due to smaller staff (eventually reaching essentially zero) and reduced tax income to the area.

8.3.3 Safe Storage-Deferred Dismantlement of the Reference Nuclear Power Plant

For this mode the reference power plant is placed in passive safe storage at the end of its operating lifetime. After a continuing care period of 50 years, the facility is dismantled. The 50-year continuing care period allows the ^{60}Co (the primary contributor to occupational exposure in the plant) to decay to very low levels.

8.3.3.1 Decommissioning Plan and Schedule of Events

A description of the activities required to place a retired facility in passive safe storage was presented in Section 8.2.2. An approximate schedule of events for safe storage of the reference power plant is presented in Figure 8.3.2. Planning and preparatory activities are carried out during the final one to two years of facility operation. Safe storage activities begin with reactor defueling and disabling operations. Spent fuel is removed to the reactor fuel storage pool and shipped offsite. The reactor coolant system, chemical volume control system, and auxiliary systems that contained reactor coolant water are drained and sealed. Selected systems may be chemically decontaminated. Extensive chemical decontamination is not required, since radiation dose levels will be decreased substantially by radioactive decay during the interim care period.

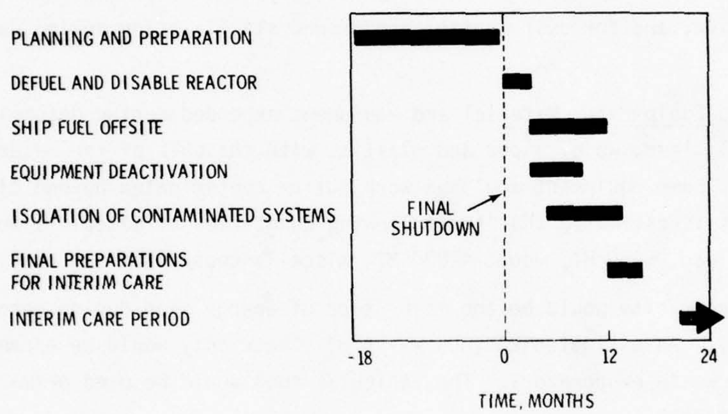

FIGURE 8.3.2. Approximate Schedule of Events for Placing the Reference Nuclear Power Plant in Protective Storage

The primary efforts during the safe storage operations are associated with deactivating reactor equipment and isolating contaminated systems. Active effluent control systems are the final equipment deactivated. Final preparations for continuing care include radiation surveys of the facility and installation of security systems and radiation detection systems to be used during the continuing care period.

A description of the activities required to dismantle a retired nuclear facility were presented in Section 8.2.1. An approximate schedule of events for dismantling the reference power plant at the end of the continuing care period is presented in Figure 8.3.3. Planning and preparation for the dismantlement takes place during the final two years of the continuing care period. The planning period concludes with a comprehensive radiation survey of the facility that serves as the basis for finalizing decommissioning plans.

The initial activity in the dismantlement is renovation of active effluent control systems and other safety-related systems that would be operated during the dismantlement. Removal, packaging, and shipping of radioactive material begins when these activities are completed in each system of the power plant.

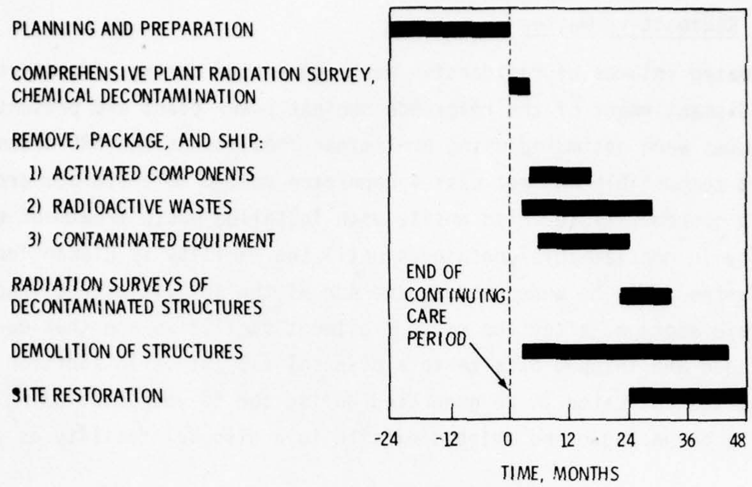

FIGURE 8.3.3. Approximate Schedule of Events for Delayed Dismantlement of the Reference Nuclear Power Plant

As much as possible, dismantlement activities are carried out in parallel in the various portions of the facility. If plant structures are to be demolished, that effort begins with the cooling tower and other nonradioactive buildings and continues as buildings are decontaminated. All structures are surveyed before demolition to assure that all radioactivity above levels permitted for unrestricted use has been removed.

8.3.3.2 Manpower Requirements

The estimated manpower requirements for safe storage decommissioning activities, 50 years of continuing care, and deferred dismantlement of the reference power plant are summarized in Table 8.3.5. These manpower estimates are based on preliminary results from the NRC decommissioning study. Manpower for demolition and site restoration is not included in the table.

TABLE 8.3.5. Estimated Manpower Requirements for Safe Storage-Deferred Dismantlement of the Reference Nuclear Power Plant

Job Description	Man-Years: Safe Storage	50-Yr Continuing Care	Deferred Dismantlement	Total
Supervisory and engineering	19	13	45	77
Operations and maintenance	10	5	31	46
Decommissioning	29	--	92	121
Health physics and safety	5	7	13	25
Radiation protection	11	14	27	52
Administrative, clerical and security	26	63	50	139
Total	100	102	258	460

8.3.3.3 Radioactive Wastes

The estimated volumes of radioactive waste generated during safe storage, continuing care, and deferred dismantlement of the reference nuclear power plant are presented in Table 8.3.6. The waste volumes were estimated using preliminary results of the NRC decommissioning study as a basis. Most combustible and wet wastes generated during safe storage are assumed to be converted to a noncombustible solid onsite with installed waste treatment equipment and stored in the facility in nonflammable containers until the facility is dismantled. About 15 m^3 of combustible wastes would be generated at the end of the activities required to place the facility in safe storage, after the waste treatment facilities are shut down. These wastes would be packaged and shipped offsite to a disposal facility. In addition, about 15 m^3 of combustible wastes estimated to be generated during the 50-year continuing care period are also assumed to be packaged and shipped offsite to a disposal facility as the wastes are generated.

Noncombustible wastes generated during dismantlement are packaged without treatment and shipped offsite. Combustible wastes are also packaged and shipped. All contaminated wastes are assumed to be shipped to shallow land burial sites. Some of the activated materials removed from the facility contain substantial amounts of radionuclides with long half lives. Under current regulations, these materials could be shipped to low-level waste burial grounds. Future regulations could require that these materials be placed in a Federal repository. The packaged wastes stored in the facility during the protective storage operations would also be shipped offsite for disposal during the dismantlement.

The general composition of the decommissioning wastes is also indicated in Table 8.3.6. Combustible wastes are generated while performing the decommissioning operations. They consist of a variety of materials including protective clothing, plastic and cloth coverings, and wood. This waste is similar to the combustible trash generated during normal plant operations and maintenance activities. Wet wastes include resins, water filters, and evaporator bottoms. Noncombustible wastes include stainless and carbon steel equipment, aluminum fuel storage racks, stainless steel pool liners, concrete rubble and HEPA filters. The estimated radioactive contamination levels on the waste are also indicated in the table. The radionuclide content of the wastes shipped at the time of dismantlement is presented in Table 8.3.7.

8.3.3.4 Routine Radioactive Effluents

Ventilation confinement during safe storage activities would be maintained on all buildings until residual contamination is removed or immobilized and all radioactive wastes are removed and packaged for disposal. Radiation areas would be sealed and provided with local filter media to allow for anticipated temperature and pressure fluctuations. No significant release of airborne radioactivity are anticipated. Any routine releases that do occur during safe storage, continuing care, or dismantlement are expected to be much smaller than releases permitted under the facility license.

TABLE 8.3.6. Estimated Waste Volumes for Safe Storage-Deferred Dismantlement of the Reference Nuclear Power Plant

Waste Type	Decommissioning Phase	Volume, m^3(a)	Waste Description	Radioactivity, Ci/m^3(b)	Waste Disposition	Time of Shipment
Noncompactable, noncombustible waste	Dismantlement	660	Activated equipment and structural materials	1×10^{2}	Packaged, shipped for disposal	Dismantlement
	Dismantlement	7,000	Contaminated equipment and structural materials	6×10^{-6}	Packaged, shipped for disposal	Dismantlement
Compactable waste and combustible trash	Safe storage	50	Combustible trash	1×10^{-6}	Treated, packaged, shipped for disposal	Dismantlement
	Safe storage	15	Combustible trash	1×10^{-2}	Packaged, shipped for disposal	Safe Storage
	Dismantlement	210	Combustible trash	1×10^{-6}	Packaged, shipped for disposal	Dismantlement
	Safe storage	30	HEPA filters	1×10^{-3}	Packaged, shipped for disposal	Dismantlement
	Dismantlement	30	HEPA filters	1×10^{-3}	Packaged, shipped for disposal	Dismantlement
	Continuing care	15	Combustible trash	1×10^{-4}	Packaged, shipped for disposal	Continuing care
Wet wastes and liquids	Safe storage	50	Spent resins	1×10^{-1}	Treated, packaged, shipped for disposal	Dismantlement
	Safe storage	10	Filter cartridges	5×10^{-2}	Treated, packaged, shipped for disposal	Dismantlement
	Safe storage	200	Evaporator bottoms	3×10^{-3}	Treated, packaged, shipped for disposal	Dismantlement

a. All volumes are for raw waste before treatment and packaging.
b. Radioactivity levels at time of shipment are given. Table 8.3.3 gives radionuclide content of wastes shipped at time of safe storage. Table 8.3.7 gives radionuclide content of wastes shipped during dismantlement.

TABLE 8.3.7. Fractional Abundance of Radionuclides in Wastes from Dismantlement of the Reference Nuclear Power Plant 50 Years After Final Reactor Shutdown

Nuclide	Half-Life, Yr	Fraction of Total Activity: Activated Material	Fraction of Total Activity: Other Wastes
^{60}Co	5.27	1.6×10^{-2}	1.0
^{55}Fe	2.7	4.2×10^{-5}	-
^{59}Ni	$\sim$80,000	7.4×10^{-3}	-
^{63}Ni	$\sim$100	.98	-

Aqueous releases during safe storage would be limited to evaporator condensates and similar liquids that would be sampled and shown to contain less radioactivity than is permitted to be released under the facility license.

8.3.3.5 Cost Estimate

The estimated costs for the safe storage, continuing care and deferred dismantlement operations are presented in Table 8.3.8. These costs were derived using assumptions consistent with the cost estimates developed for a recent fuel reprocessing plant decommissioning study.(7) The allowance for utilities is primarily to cover the cost of electricity during the decommissioning operation. Equipment costs include specialized decommissioning equipment that must be purchased as well as expendable equipment and supplies. Waste packaging costs, environmental surveillance costs, and specialized sub-contractor costs are included in supporting activities. The cost of waste transportation and disposal is not included.

TABLE 8.3.8. Estimated Costs for Safe Storage-Deferred Dismantlement of the Reference Nuclear Power Plant

	Cost, 1000s of Mid-1976 Dollars			
Cost Element	Safe Storage	50 Yr Continuing Care	Deferred Dismantlement	Total
Labor	2,000	460	4,400	6,900
Utilities	1,500	250	720	2,500
Equipment and supplies	100	150	320	570
Supportive activities	100	1,800	--	1,900
Demolition and site resotration	NA	NA	4,700	4,700
Owner costs	1,100	710	2,500	4,300
Contingency	1,000	710	2,500	4,300
Total	5,900	4,100	15,000	25,000

8.3.3.6 Nonradiological Impacts

The nonradiological impacts of decommissioning the reference power plant are discussed briefly below. Where it has not been possible to quantify the impacts, they have been related to similar impacts during construction or operation of the facility. In general, the decommissioning impacts are expected to be less than similar impacts during construction and operation.

Land Use. The total site would continue to be restricted until the safe storage decommissioning activities are completed. At the start of the continuing care period, the perimeter fence would be moved to encompass a total area of about 81,000 m^2 (20 acres). This area would be restricted for the 50-year continuing care period. When all decontamination and dismantlement work is completed and the facility license is terminated, unrestricted use of the entire site would be possible. Land would be required at a commercial burial ground and possibly in a Federal repository to dispose of the radioactive decommissioning wastes.

Water Use Water usage during decommissioning activities would be reduced from operational conditions. Principal use of water is sanitary systems, flushing, and surface washing during decontamination, and for dust control and ground stabilization during demolition and site restoration.

Material and Equipment. Material and equipment expended during both the safe storage decommissioning activities and the final dismantlement would be principally steel, lead, wood, paper and plastic, with the bulk of material used for shipping containers during the final dismantlement phase. Some equipment would be worn out or contaminated beyond cleaning and would be discarded. It is estimated that the following total quantities of materials would be expended: steel, $\sim$75 MT; lead, $\sim$50 MT; wood, $\sim$2000 MT; and miscellaneous paper and plastics, $\sim$40 MT.

Energy. Electricity would be the major type of energy used during decommissioning, followed by vehicular fuel. An estimated 120,000 MW-hr of electricity would be expended, largely in operating the radwaste evaporators during the protective storage decommissioning activities. An estimated 60,000 MW-hr of electricity would be expended for the final dismantlement. The vehicular fuel would be used principally during demolition and site restoration.

Air Quality Effects. No significant effects are anticipated during the safe storage, continuing care, or final dismantlement periods. Any effects of vehicular exhaust gases or of fugitive dust generated from building demolition, loading of concrete rubble or grading and backfilling operations would be transitory. Areas to be graded or excavated would usually be wet down to minimize dust generation.

Noise Impacts. No significant noise impacts are expected. Onsite noise levels during safe storage are likely to be smaller than during facility operation. During dismantlement, the noisiest work phase would be the demolition and site restoration. Noise levels during these operations are expected to be comparable to noise levels during facility construction.

Effects on Nearby Communities. The principal impacts on the local communities would be reduced payroll due to smaller staff and reduced tax income to the area.

8.3.4 Selection of a Decommissioning Alternative for the Reference Nuclear Power Plant

The safe storage-deferred dismantlement mode was selected as the reference alternative for decommissioning the reference nuclear power plant. The radioactive decay in the facility over the 50-year continuing care period significantly reduces radiation levels in the facility. This simplifies many of the dismantlement techniques, resulting in lower costs than immediate dismantlement. The radioactive decay also significantly reduces occupational exposure during the dismantlement. The total volume of waste requiring disposal from dismantlement 50 years after reactor shutdown is also less than for immediate dismantlement by 3800 m^3. This results because some wastes that required offsite disposal for immediate dismantlement can be decontaminated to releasable radiation levels after the waste has decayed for 50 years.

8.4 DECOMMISSIONING OF AN INDEPENDENT SPENT FUEL STORAGE FACILITY

8.4 DECOMMISSIONING OF AN INDEPENDENT SPENT FUEL STORAGE FACILITY

Two decommissioning modes were considered for an independent spent fuel storage facility (ISFSF). These are 1) immediate dismantlement and 2) hardened safe storage. These decommissioning modes were considered because the ISFSF is expected to contain modest amounts of radioactivity at shutdown and because the residual radioactivity is not expected to contain significant amounts of long half-life radioisotopes.

This analysis assumes that facility shutdown activities are complete when decommissioning begins. Activities assumed to have been carried out to shut down the facility include: removal of all spent fuel stored in the facility; processing, packaging and removal of all radioactive wastes generated during facility operations or shutdown activities; and removal from the site of hazardous chemicals or flammable materials not required for the decommissioning operations.

8.4.1 Independent Spent Fuel Storage Facility Description

The essential features of the reference 3000 MTHM independent spent fuel storage facility are presented in Section 3.2. In addition to the portions of the facility described in that section, the ISFSF is also assumed to contain facilities for immobilization of radioactive liquids and wet wastes and incineration of combustible radioactive wastes.

To identify the potential environmental effects of decommissioning the reference ISFSF, the amount of residual radioactivity in the facility at shutdown was estimated (Table 8.4.1). The decay of these materials after shutdown is also shown. The table is based on the assumption that 2% of the activity in column 6 of reference Table 3.3.4 is accumulated each year over the 30-year lifetime of the plant. The radioactivity contained in HEPA filters at shutdown is not included in the table. This radioactivity is estimated to be about 500 Ci of the radioisotopes listed in column 6 of reference Table 3.3.4.

8.4.2 Immediate Dismantlement of the Reference Independent Spent Fuel Storage Facility

For the dismantlement mode, all radioactive materials are removed from the facility to an approved disposal site. At the discretion of the facility owner, buildings that contained radioactive material may be demolished and the site restored to approximately its pre-facility condition.

8.4.2.1 Decommissioning Plan and Schedule of Events

A description of the general types of activities carried out to dismantle a retired facility was presented in Section 8.2.1. An approximate schedule of events for dismantlement of the ISFSF is presented in Figure 8.4.1. The planning and preparatory activities are carried out during the final one to two years of facility operations. After facility shutdown activities have been completed, the fuel storage pool water is treated to remove as much residual contamination as practicable and the pools are drained. Some of the pool water may be evaporated and released to the atmosphere. Most of the water is assumed to be cleaned to

TABLE 8.4.1. Estimated Inventory of Radioactive Materials in Independent Spent Fuel Storage Facility at Final Shutdown[(a)]

Isotope	Radioactivity, Ci Shutdown	10 Yr	20 Yr	30 Yr	50 Yr	100 Yr
Activation Products						
^{51}Cr	2.0×10^{-3}	0.0	0.0	0.0	0.0	0.0
^{54}Mn	2.0×10^{-1}	6.6×10^{-5}	2.2×10^{-8}	0.0	0.0	0.0
^{55}Fe	1.3×10^{1}	1.0	8.0×10^{-2}	1.2×10^{-2}	3.6×10^{-5}	0.0
^{59}Fe	1.8×10^{-2}	0.0	0.0	0.0	0.0	0.0
^{58}Co	6.8×10^{-1}	0.0	0.0	0.0	0.0	0.0
^{60}Co	1.9×10^{1}	5.0	1.3	3.4×10^{-1}	2.2×10^{-2}	3.4×10^{-5}
Subtotal	3.3×10^{1}	6.0	1.4	3.5×10^{-1}	2.2×10^{-2}	3.4×10^{-5}
Fission Products						
^{89}Sr	9.4×10^{-2}	0.0	0.0	0.0	0.0	0.0
^{90}Sr	7.2	5.6	4.4	3.6	2.2	6.6×10^{-1}
^{95}Zr	4.2×10^{-3}	0.0	0.0	0.0	0.0	0.0
^{103}Ru	3.6×10^{-4}	0.0	0.0	0.0	0.0	0.0
^{106}Ru, ^{106}Rh[(b)]	3.2×10^{-2}	3.4×10^{-5}	3.6×10^{-8}	0.0	0.0	0.0
^{127m}Te	4.8×10^{-2}	0.0	0.0	0.0	0.0	0.0
^{129m}Te	4.6×10^{-3}	0.0	0.0	0.0	0.0	0.0
^{134}Cs	4.1×10^{1}	1.5	4.8×10^{-2}	1.6×10^{-3}	2.0×10^{-7}	0.0
^{137}Cs	4.3×10^{2}	3.3×10^{2}	2.7×10^{2}	2.0×10^{2}	1.3×10^{2}	4.2×10^{1}
^{141}Ce	4.4×10^{-4}	0.0	0.0	0.0	0.0	0.0
^{144}Ce, ^{144}Pr[(b)]	6.0×10^{-2}	8.2×10^{-6}	5.7×10^{-10}	0.0	0.0	0.0
Subtotal	4.7×10^{2}	3.3×10^{2}	2.7×10^{2}	2.0×10^{2}	1.3×10^{2}	4.2×10^{1}
Total	5.0×10^{2}	3.3×10^{2}	2.7×10^{2}	2.0×10^{2}	1.3×10^{2}	4.2×10^{1}

a. Does not include radioactive materials in HEPA filters. Radioactivity shown is based on an accumulation of 2% per year of the radioactive materials in Reference Table 3.3.4, Column 6, over the 30-year lifetime of the plant.
b. Total activity (parent plus daughter) listed.

levels that are consistent with ALARA (as low as reasonably achievable) practices. These levels are expected to be well below those permitted in Federal regulations for release to local water bodies.[(1)] After the pools are drained, the contaminated equipment is removed from the pools and water treatment areas, and structural surfaces in these areas are decontaminated. Contamination is then removed from the waste treatment areas.

The final operation in the facility involving radioactive materials is the removal of equipment and structural decontamination of the ventilation system. The decontaminated structures may then be demolished and the site restored to approximately its prefacility condition. A final radiation survey ensures that all potentially hazardous amounts of radioactivity have been removed from the site.

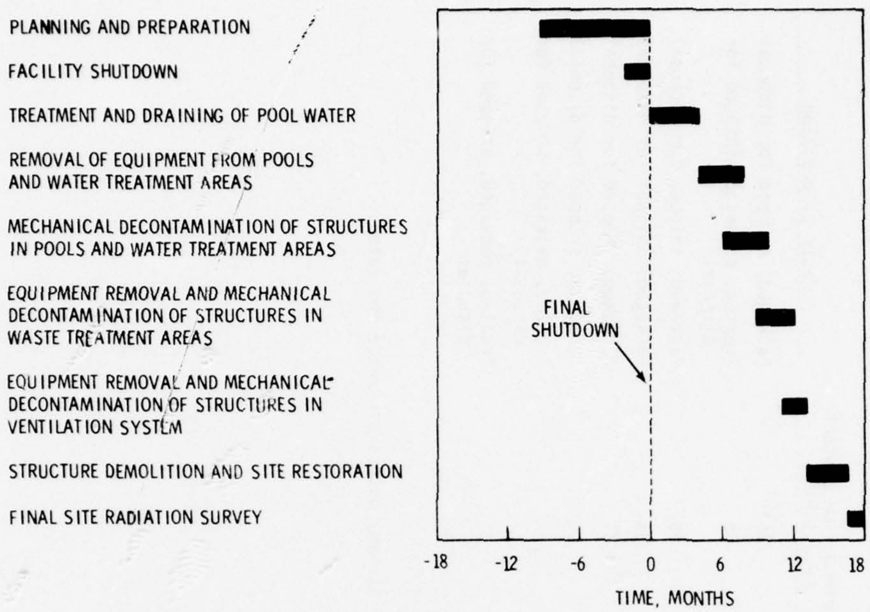

FIGURE 8.4.1. Approximate Schedule of Events for Dismantling the Reference Independent Spent Fuel Storage Facility

8.4.2.2 Manpower Requirements

The estimated manpower requirements for dismantlement of the ISFSF are presented in Table 8.4.2. These manpower estimates are based on assumptions for staffing requirements consistent with those used to derive the manpower estimate for decommissioning the reference fuel reprocessing plant. Manpower for demolition and site restoration is not included in the table.

TABLE 8.4.2. Estimated Manpower Requirements for Dismantlement of the Reference Independent Spent Fuel Storage Facility

Job Description	Man-years
Facility operations and maintenance	15
Engineering and supervisory	15
Health physics and laboratory	4
Radiation protection	6
Skilled decommissioning	25
Management, clerical and security	30
Total	95

8.4.2.3 Radioactive Wastes

The estimated volumes of radioactive waste generated from dismantlement of the reference ISFSF are presented in Table 8.4.3. Estimates of radioactive contamination levels in the waste are also presented in the table.

TABLE 8.4.3 Estimated Waste Volumes from Immediate Dismantlement of the Reference Independent Spent Fuel Storage Facility

Waste Type	Volume, m^3[a]	Waste Description	Radionuclide Content Fraction[b]	Waste Disposition
Compactible waste and combustible trash	30	HEPA filters	.25[c]	Packaged, shipped for disposal
	100	Combustible trash	.05	Treated, packaged, shipped for disposal
	20	Combustible trash	.001	Packaged, shipped for disposal
Noncompactible, noncombustible wastes	4,300	Equipment and structural materials	.36	Packaged, shipped for disposal
	120	Equipment	.04	Packaged, shipped for disposal
	60	Equipment and structural materials	.05	Packaged, shipped for disposal
Wet wastes and liquids	20	Combustible wet wastes	.2	Treated, packaged, shipped for disposal
	30	Noncombustible wet wastes	.3	Treated, packaged, shipped for disposal

a. All volumes are raw waste before treatment or packaging.
b. Fraction of total activity in shutdown column of Table 8.4.1 present in this waste stream, unless otherwise indicated.
c. Reference Table 3.3.4, column 6.

All wet wastes and most combustible wastes are assumed to be treated onsite with installed waste treatment equipment. About 20 m^3 of combustible wastes are estimated to be produced after the installed waste treatment equipment is shut down. This waste would be packaged and shipped offsite for disposal without further treatment. All wastes are assumed to be packaged and shipped to an approved disposal site for nontransuranic wastes.

The general composition of the wastes is also indicated in the table. Combustible wastes consist of a variety of materials including protective clothing, plastic and cloth coverings, rags, paper, and wood. The combustible wastes generated during decommissioning are expected to be similar to the combustible trash generated during plant operations and maintenance activities. Wet wastes include filter cartridges, spent resins, slurrys, and concentrated liquids from the pool water treatment system. These wastes are also expected to be similar in composition to operating wastes. Noncombustible wastes include HEPA filters, aluminum and stainless steel equipment, concrete rubble, and stainless steel pool liners and ventilation ductwork.

8.4.2.4 Routine Radioactive Effluents

Routine radioactive releases during dismantlement of the ISFSF are expected to be aqueous releases from draining the fuel storage pools and atmospheric releases of gases and particulates through the facility ventilation system. About 18,000 m^3 of slightly contaminated water is expected to be released to local water bodies when the fuel storage pools are drained. Radioactive contamination in this water is estimated to be 5×10^{-4} times the shutdown column of Table 8.4.1.

A variety of decommissioning operations can cause radioactive materials to become airborne inside the facility. These include spraying of decontamination solutions, cutting aluminum equipment and stainless steel pool liners and equipment, and removal of contaminated concrete. Procedures are followed to reduce the amount of these airborne materials that reach the ventilation system. Nevertheless, a small fraction of the airborne radioactive materials can be expected to be released to the atmosphere through the facility stack after the materials pass through the filtered ventilation system. The amount of material released in this manner is estimated to be 10^{-9} times the shutdown column of Table 8.4.1.

8.4.2.5 Cost Estimate

The estimated costs for dismantling the reference ISFSF are presented in Table 8.4.4. These costs were derived using assumptions consistent with the cost estimates developed in a recent fuel reprocessing plant decommissioning study.[(7)] The procedures outlined in this study have assumed that the dismantlement proceeds with a minimum of difficulties. A 25% contingency is included in the cost estimate to account for the costs of dealing with unforeseen circumstances. Costs are presented in mid-1976 dollars.

Labor costs indicated in the table include direct wages, fringe benefits and other direct employer expenses. The allowances for utilities and services is to cover the costs of electricity, boiler fuels, water and water treatment, sewage treatment or disposal, and other utilities and services required during the decommissioning operation. Equipment costs include development and design costs for specialized decommissioning equipment and allowances for other

TABLE 8.4.4. Estimated Costs for Immediate Dismantlement of the Reference Independent Spent Fuel Storage Facility

Cost Element	Cost, 1000s of mid-1976 dollars
Labor	2400
Equipment and supplies	500
Utilities and services	300
Demolition and site restoration	300
Owner costs	800
Contingency	800
Total	5000

equipment that must be purchased, as well as expendable equipment and supplies and wastes packaging. Owner costs include expenses incurred directly by the owner during planning and performance of the decommissioning operations. These costs include overheads such as engineering, operating and accounting expenses, insurance, taxes, permits, licenses, and the cost of making reports and applications to regulatory agencies. The cost of radioactive waste transportation and disposal is not included in the cost estimate.

8.4.2.6 Nonradiological Impacts

The nonradiological impacts of dismantling the reference ISFSF are discussed briefly below. Where it has not been possible to quantify the impacts, they have been related to similar impacts during construction or operation of the facility. In general, the decommissioning impacts are expected to be less than similar impacts during construction and operation.

Land Use. The facility site (4 km^2) is maintained during the time that the dismantling operation is being carried out. When the dismantlement is completed and the facility license is terminated, the site may be released for alternative uses as the owner desires. Land will be required at a commercial burial ground to dispose of the radioactive decommissioning wastes.

Water Use. Requirements for water during decommissioning are reduced from requirements during plant operations. Sanitary water usage is less since the decommissioning work force is less than the work force during plant operations. Requirements for storage pool cooling water and process water are virtually eliminated after the plant is shut down (except for water used in the waste treatment systems).

Equipment and Materials. Some materials and equipment used for decommissioning become radioactively contaminated and are removed from further use following the decommissioning operations. These include steel radioactive waste disposal containers; cloth, paper, plastic, and wood used for personnel protection and contamination control; and specialized decommissioning equipment such as cutting torches, hand tools, rock splitters and so forth. The approximate amounts of material used for this decommissioning mode are: steel shipping containers, 800 MT; paper, wood, plastic, 30 MT; and equipment (mostly steel), 80 MT.

Energy. Electricity usage during decommissioning should be less than during plant operations. The primary use of electricity during decommissioning is operation of the ventilation system (about 5 MW-hr for dismantlment). Plant vehicles and equipment used during decommissioning will burn diesel fuel and gasoline. Heavy equipment used during the demolition and site restoration phase of dismantlement will be the primary consumer of vehicular fuels.

Air Quality Effects. During the demolition and site restoration phase of dismantlement, quantities of fugitive dust will be generated from building demolition, loading of concrete rubble, grading and backfilling operations. Vehicle effluents will also be produced. The effects would be transitory and will be confined to the immediate vicinity of the site. Areas to be graded or excavated will usually be wet down to minimize dust generation.

Noise Impacts. Noise caused by decommissioning will vary with day-to-day work schedules, weather conditions, and other factors. The noisiest phase of dismantlement is expected to be the demolition and site restoration phase (about six months total time). Noise levels during these operations are expected to be comparable to the noise levels during facility construction.

Effects on Nearby Communities. The major impact on local communities would be associated with reduction in the work force during the decommissioning. The work force at the plant would be reduced from about 100 people during normal operations to about 50 during the dismantlement. The work force would be gradually reduced to zero at the end of the dismantlement.

8.4.3 Hardened Safe Storage of the Reference Independent Spent Fuel Storage Facility

To place the reference ISFSF in hardened safe storage, all residual radioactivity in the facility is confined to portions of the plant that are separated from the remainder of the facility and the environment by permanent physical barriers. For this study, it is assumed that the facility is maintained in this condition until all radioactivity has decayed to very low levels. It is estimated that the continuing care period will last about 100 years. Activities will be required at the end of the continuing care period to terminate the facility license. These activities will include performance of radiation surveys and removal of any residual radioactive contamination above limits established by regulatory agencies for public access to the facility.

8.4.3.1 Decommissioning Plan and Schedule of Events

A description of the general types of activities required to place a retired nuclear facility in hardened safe storage is presented in Section 8.2.2. An approximate schedule of events for hardened safe storage of the reference ISFSF is presented in Figure 8.4.2. Planning and preparation takes place during the final one to two years of facility operations. After facility shutdown activities are completed, the fuel storage pool water is treated to remove as much residual contamination as practicable and the pools are drained, as for dismantlement. Equipment in the facility is deactivated and areas outside the area to be isolated are decontaminated. Contaminated equipment and materials outside the area to be isolated are placed in the storage pools.

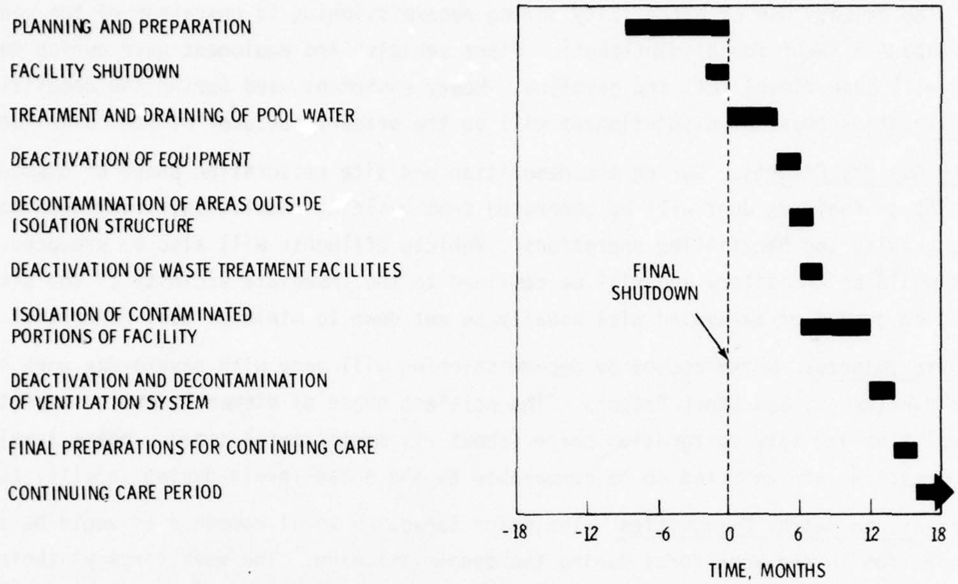

FIGURE 8.4.2. Approximate Schedule of Events for Hardened Safe Storage of the Independent Spent Fuel Storage Facility

Following deactivation of the waste treatment equipment, the storage pools and waste treatment areas are separated from the remainder of the facility and the environment by permanent physical barriers. Any radioactive material remaining outside the isolation structure is removed, packaged, and shipped offsite.

The ventilation system is the final portion of the facility to be deactivated. All radioactively contaminated equipment, filters, and ductwork are packaged and shipped offsite for disposal. Most exterior doors into the facility are disabled. The exclusion area fence is maintained and the remainder of the site is assumed to be released for alternate uses. Activities during the continuing care period and the final decommissioning activities include environmental monitoring and inspection of the isolation structures. The facility is protected from unauthorized entry by the exclusion fence and by locked or disabled exterior doors.

8.4.3.2 Manpower Requirements

The estimated manpower requirements for hardened safe storage of the reference ISFSF are presented in Table 8.4.5. The manpower requirements for one year of continuing care are also given in the table. Continuing care activities are assumed to be carried out for 100 years or more.

TABLE 8.4.5. Estimated Manpower Requirements for Hardened Safe Storage of the Independent Spent Fuel Storage Facility

Job Description	Man-years Safe Storage	Man-years Continuing Care(a)	Final Decommissioning
Facility operation and maintenance	10		
Engineering and supervisory	10	0.1	0.5
Health physics and laboratory	3		0.5
Radiation protection	4	0.1	1.5
Skilled decommissioning	15		2.0
Management, clerical and security	20	0.1	0.5
Total	60	0.3	5.0

a. Continuing care activities are carried out for 100 years or more. Manpower requirements per year are shown.

8.4.3.3 Radioactive Wastes

Most of the radioactivity in the ISFSF at shutdown remains in the isolated portion of the facility. However, there are some wastes generated during the safe storage operations that will require offsite disposal. These wastes are composed primarily of waste generated in the final cleanup operations after the isolation structures are closed and also composed of combustible wastes generated after the onsite combustible waste treatment facility is decommissioned. Combustible and wet wastes generated during the first phases of the decommissioning operation are assumed to be converted to a nonflammable solid in the installed waste treatment facilities and placed in the isolation structure in nonflammable containers. The estimated volumes of waste requiring offsite disposal during the safe storage operation are summarized in Table 8.4.6. Less than one m^3/yr of radioactive wastes are estimated to be generated during the continuing care period.

TABLE 8.4.6. Estimated Waste Volumes for Hardened Safe Storage of the Reference Independent Spent Fuel Storage Facility

Waste Type	Volume, m^3(a)	Waste Description	Radionuclide Content Fraction(b)	Waste Disposition
Compactible waste and combustible trash	30	HEPA filters	.25(c)	Packaged, shipped for disposal
	10	Combustible trash	.001	Packaged, shipped for disposal
Noncompactible, noncombustible wastes	160	Equipment and Structure Materials	.01	Packaged, shipped for disposal

a. All volumes are raw waste before treatment or packaging.
b. Fraction of total activity shown in shutdown column of Table 8.4.1 in this waste stream unless otherwise indicated.
c. Reference Table 3.3.4, Column 6.

Combustible wastes generated during the latter stages of the safe storage operation are typical of combustible wastes generated during plant operations and maintenance activities. They are composed of protective clothing, plastic and cloth coverings, and wood.

Noncombustible wastes shipped offsite during the safe storage operations are primarily produced from the decontamination of the ventilation system. They include HEPA filters, stainless steel ductwork and equipment, and concrete rubble.

Estimated radioactive contamination levels of the wastes are also included in the table. Radioactive contamination levels on the waste are expected to be typical of operating wastes from the facility.

8.4.3.4 Routine Radioactive Effluents

Routine radioactive releases in preparation for or during safe storage are expected to be aqueous releases from draining the fuel storage pools and atmospheric releases of gases and particles through the facility ventilation system. About 18,000 m^3 of slightly contaminated water is expected to be released to local water bodies when the fuel storage pools are drained. Radioactive contamination in this water is estimated to be 5×10^{-4} times the shutdown column of Table 8.4.1.

Airborne releases are expected to be less than releases during dismantlement, since fewer decommissioning activities are carried out that could cause radioactive material to become airborne in the facility. Contamination control methods would be utilized for decommissioning operations that could result in airborne contamination inside the facility. Based on the release predicted for dismantlement it is estimated that the routine airborne radioactive effluent from entombing the reference ISFSF will be 10^{-11} times the shutdown column of Table 8.4.1.

8.4.3.5 Cost Estimate

The estimated costs for placing the reference ISFSF in hardened safe storage are presented in Table 8.4.7. The assumptions, definitions, and limitations associated with this cost estimate are similar to those presented for the dismantlement mode. The estimated annual cost of interim care and the final decommissioning costs are also indicated in the table. Most of the cost of continuing care is associated with owner costs such as taxes and insurance. Continuing care activities at an ISFSF placed in hardened safe storage are assumed to be carried out for 100 years or more. Waste transportation and disposal costs are not included.

TABLE 8.4.7. Estimated Costs for Hardened Safe Storage of the Reference Independent Spent Fuel Storage Facility

Cost Element	Cost, 1000s of Mid-1976 Dollars: Safe Storage	Cost, 1000s of Mid-1976 Dollars: Continuing Care, Per Year	Final Decommissioning
Labor	1,500	8	125
Equipment and supplies	500	5	15
Utilities and services	200	2	2
Owner costs	500	10	2
Contingency	500	5	36
Total	3,000	30	180

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8.4.3.6 Nonradiological Impacts

The nonradiological impacts of the hardened safe storage operations for the reference ISFSF are discussed briefly below. Where it was not possible to quantify the impacts, they were related to similar impacts during construction or operation of the facility. In general, the decommissioning impacts are expected to be less than similar impacts during construction or normal operation of the facility.

Land Use. The 0.1 km^2 (30 acre) exclusion area of the facility site is occupied for 100 years by the facility. The remaining 4 km^2 of the site could be released for alternate use at the discretion of the facility owner. Activities near the exclusion area are restricted to those that would not jeopardize the integrity of the entombed structure. Some land is required at a commercial burial ground to dispose of the radioactive decommissioning wastes.

Water Use. Requirements for water during decommissioning are reduced from requirements during plant operations. Sanitary water usage is less since the decommissioning work force is less than the work force during plant operations. Requirements for pool cooling water and process water are virtually eliminated after the plant is shut down (except for water used in the waste treatment systems).

Equipment and Materials. Some materials and equipment used for decommissioning become radioactively contaminated and are removed from further use following the decommissioning operations. These include steel radioactive waste disposal containers, cloth, paper, plastic and wood used for personnel protection and contamination control, and specialized decommissioning equipment such as cutting torches, temporary shielding, hand tools, rock splitters and so forth. The approximate amounts of material used for this decommissioning mode is as follows: steel shipping containers, 50 MT; paper, wood, and plastic, 20 MT; and equipment (mostly steel), 5 MT.

Energy. Electricity used during decommissioning should be less than during plant operations. The primary use of electricity is for operating the ventilation system (about 3 MW-hr for safe storage). Diesel fuel and gasoline would be burned in plant vehicles and equipment used during decommissioning. Fossil fuels may be consumed for space heating. The amount of vehicle and other fossil fuels consumed during decommissioning is expected to be less than the amounts consumed during plant operations.

Air Quality Effects. The safe storage operations are not expected to noticeably affect air quality near the site. The primary source of effluents would be from vehicular traffic. Vehicular traffic during the decommissioning operations is expected to be less than the traffic during plant operations.

Noise Impacts. Noise caused by decommissioning would vary with day-to-day work schedules, weather conditions, and other factors. The primary source of noise during the safe storage operations would be vehicular traffic; however, vehicular traffic is expected to be less than during plant operations.

Effects on Nearby Communities. The major impact on local communities would be associated with reduction in the work force at the plant. The work force would be reduced from about

100 people during normal operations to about 40 people during the decommissioning operations to the equivalent of less than one person during the continuing care period.

Institutional Effects. The continuing care activities at the facility must continue for 100 years or more. Institutions for assuring that the isolation structure remains in a condition that poses minimum risks to the public and the environment for these long time periods do not currently exist.

8.4.4 Selection of a Decommissioning Alternative for the Reference Independent Spent Fuel Storage Facility

Immediate dismantlement was selected as the reference alternative for decommissioning the reference ISFSF. Although residual radiation levels in the facility after shutdown are expected to be low, hardened safe storage of the facility would require surveillance activities at the site for 100 years or more to assure public protection from the residual radioactivity. Corporate survivability of the facility owner or assumption of the surveillance requirements by a government organization would be required to assure that the surveillance program would be carried out for the required period. The institutions to guarantee that surveillance requirements would be completed are not currently in place. Immediate dismantlement does not require such a surveillance program. Dismantlement also releases the site for other uses and assures that the facility owner pays all decommissioning costs. The cost of dismantlement of the ISFSF is only moderately more expensive than hardened safe storage.

8.5 DECOMMISSIONING OF A FUEL REPROCESSING PLANT

8.5 DECOMMISSIONING OF A FUEL REPROCESSING PLANT

Three potential decommissioning alternatives for a fuel reprocessing plant (FRP) were considered. These alternatives include: 1) immediate dismantlement, 2) hardened safe storage with an extended continuing care period, and 3) safe storage followed by dismantlement after a suitable period to permit some decay of residual radioactivity in the facility. The safe storage-dismantlement and hardened safe storage modes were chosen for detailed consideration. Results from a current decommissioning study[7] indicate that the present value cost of the safe storage-dismantlement is less than the cost of immediate dismantlement. Occupational exposure is also somewhat greater for immediate dismantlement. Other potential environmental effects for the two modes are expected to be nearly equal. The current study indicates that economic and occupational exposure considerations favor deferring dismantlement about 30 years after facility shutdown. This time period is used in this assessment for the safe storage-dismantlement mode. Hardened safe storage is considered for comparison, even though it may not be a viable decommissioning option for a fuel reprocessing plant. Activities would be required at the end of the continuing care period following hardened safe storage to finally decommission the facility and terminate the facility license.

The analysis presented here assumes that facility shutdown activities have been completed when decommissioning begins. Activities assumed to have been carried out to shut down the facility include: product recovery flushes; processing of all inventories of high, intermediate, and low-level liquid wastes through the installed waste treatment facilities; removal of all product materials from the facility; removal of all inventories of SHLW and packaged cladding residue; processing, packaging and removal of all other radioactive wastes generated during plant operation or shutdown activities; removal from the site of bulk quantities of process chemicals; and completion of preparations to close out special nuclear material accountability requirements.

8.5.1 Fuel Reprocessing Plant Description

The essential features of the reference 2000-MTHM/yr fuel reprocessing plant are presented in Section 3.2. However, other assumptions were necessary to estimate the cost and identify the potential environmental effects of decommissioning an FRP. These assumptions are outlined below:

- All equipment in the shearing, dissolution, feed preparation, codecontamination, high-level liquid waste concentration and storage and high-level liquid waste solidification portions of the fuel reprocessing cycle is operated and maintained remotely. The remainder of the plant equipment is maintained by contact methods.
- Facilities are installed in the plant for remote chemical decontamination of equipment and structural surfaces in the contact maintenance areas.
- A special cell is provided for remote repair, decontamination and/or disassembly of equipment removed from the process cells.

- A high-level liquid waste solidification facility similar to the one described in Section 4.1 is installed in the facility.
- Underwater storage pools for the solidified high-level waste containers similar to those described in Section 5.4.1 are installed. There is sufficient storage available for five years' output from the high-level waste solidification facility. Facilities to load and ship the SHLW containers are provided.
- Low-level radioactive liquids and wet waste are immobilized in an installed cementation (or bituminization) facility before being shipped offsite for disposal.
- Intermediate-level liquid wastes are concentrated and combined with the low-level liquid wastes for immobilization and disposal. Tank storage for 450,000 ℓ of intermediate-level liquid wastes is provided in the facility.
- Storage capacity is provided in the facility for treated, packaged non-high-level waste awaiting shipment. Loadout facilities for packaged waste are included.
- An incinerator is installed in the facility for treating combustible wastes. Combustible TRU waste ash is processed through the waste immobilization system before being shipped for disposal.
- Cladding hulls are mixed with sand and sealed in 75 cm dia x 2.2 m long canisters. A separate facility for onsite storage of cladding hulls was not considered in the decommissioning analysis.
- Process vessel off-gas treatment equipment is installed. Iodine, carbon-14, and krypton are removed from the off-gas. Most of the tritium, is released to the atmosphere through the plant stack.

It is necessary to estimate the amount of radioactive material remaining in the plant when decommissioning begins to identify the potential environmental effects of decommissioning the reference FRP. This estimate is presented in the shutdown column of Table 8.5.1.

The table also shows the decay of the residual radioactivity following facility shutdown. The activities presented in the table are estimates of the activity remaining after completion of predecommissioning chemical decontamination procedures. The estimates are based on an accumulation of about 2 kg of spent fuel products in the facility each year over a 30-yr plant lifetime. Spent fuel characteristic of that to be reprocessed in the year 2000 is used as the feed to the plant. The fuel is assumed to be reprocessed 1.5 years after reactor discharge.

8.5.2 Safe Storage-Dismantlement of the Reference Fuel Reprocessing Plant

In the safe storage-dismantlement decommissioning mode the reference FRP is placed in passive safe storage at the end of its operating lifetime. After a continuing care period of 30 years, the facility is dismantled and all hazardous materials are removed from the site.

TABLE 8.5.1. Estimated Inventory of Radioactive Material in the Reference Fuel Reprocessing Plant at Final Shutdown and at Various Periods Following Shutdown[(a)]

Fission Products	Radioactivity (Ci) Shutdown	10 Yr	20 Yr	30 Yr	100 Yr	500 Yr	1000 Yr
^{79}Se	2.0×10^{-2}	2.0×10^{-2}	2.0×10^{-2}	2.0×10^{-2}	2.0×10^{-2}	2.0×10^{-2}	2.0×10^{-2}
^{87}Rb	9.6×10^{-7}	9.6×10^{-7}	9.6×10^{-7}	9.6×10^{-7}	9.6×10^{-7}	9.6×10^{-7}	9.6×10^{-7}
^{90}Sr, ^{90}Y	5.2×10^{3}	4.0×10^{3}	3.1×10^{3}	2.5×10^{3}	4.4×10^{-2}	2.3×10^{-2}	1.0×10^{-7}
^{93}Zr	9.6×10^{-2}	9.6×10^{-2}	9.6×10^{-2}	9.6×10^{-2}	9.6×10^{-2}	9.6×10^{-2}	9.6×10^{-2}
^{93m}Nb	5.4×10^{-2}	7.1×10^{-2}	8.1×10^{-2}	8.6×10^{-2}	9.6×10^{-2}	9.6×10^{-2}	9.6×10^{-2}
^{106}Ru, ^{106}Rh[(b)]	1.1×10^{3}	1.1	1.1×10^{-3}	1.0×10^{-6}	0.0	0.0	0.0
^{107}Pd	7.1×10^{-3}	7.1×10^{-3}	7.1×10^{-3}	7.1×10^{-3}	7.1×10^{-3}	7.1×10^{-3}	7.1×10^{-3}
^{110m}Ag	1.6	7.2×10^{-5}	3.4×10^{-9}	1.5×10^{-13}	0.0	0.0	0.0
^{110}Ag	2.1×10^{-1}	9.6×10^{-6}	4.3×10^{-10}	1.9×10^{-14}	0.0	0.0	0.0
^{113m}Cd	3.2×10^{-1}	2.0×10^{-1}	1.2×10^{-1}	7.1×10^{-2}	2.3×10^{-3}	5.8×10^{-12}	0.0
^{119m}Sn	6.6×10^{-3}	2.6×10^{-7}	1.0×10^{-11}	0.0	0.0	0.0	0.0
^{121m}Sn	2.6×10^{-5}	2.4×10^{-5}	2.2×10^{-5}	2.0×10^{-5}	1.1×10^{-5}	2.7×10^{-7}	2.9×10^{-9}
^{125}Sb	4.6×10^{1}	3.5	2.7×10^{-1}	1.9×10^{-2}	3.3×10^{-10}	0.0	0.0
^{126}Sn, ^{126m}Sb[(b)]	6.5×10^{-2}	6.6×10^{-2}	6.6×10^{-2}	6.6×10^{-2}	6.6×10^{-2}	6.6×10^{-2}	6.6×10^{-2}
^{126}Sb	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}
^{129}I	2.1×10^{-4}	2.1×10^{-4}	2.1×10^{-4}	2.1×10^{-4}	2.1×10^{-4}	2.1×10^{-4}	2.1×10^{-4}
^{134}Cs	7.1×10^{2}	2.4×10^{1}	8.2×10^{-1}	2.8×10^{-2}	1.5×10^{-12}	0.0	0.0
^{135}Cs	1.9×10^{-2}	1.9×10^{-2}	1.9×10^{-2}	1.9×10^{-2}	1.9×10^{-2}	1.9×10^{-2}	1.9×10^{-2}
^{137}Cs	4.0×10^{3}	3.2×10^{3}	2.5×10^{3}	2.0×10^{3}	4.0×10^{2}	3.9×10^{-2}	3.7×10^{-7}
^{137m}Ba	3.8×10^{3}	3.0×10^{3}	2.4×10^{3}	1.9×10^{3}	3.7×10^{2}	3.6×10^{-2}	3.5×10^{-7}
^{144}Ce, ^{144}Pr[(b)]	1.1×10^{3}	1.5×10^{-1}	2.0×10^{-5}	2.8×10^{-9}	0.0	0.0	0.0
^{147}Pm	5.2×10^{2}	3.7×10^{1}	2.6	1.2×10^{-1}	1.7×10^{-9}	0.0	0.0
^{151}Sm	6.4×10^{1}	5.9×10^{1}	5.5×10^{1}	5.1×10^{1}	2.9×10^{1}	1.2	2.2×10^{-2}
^{152}Eu	4.4×10^{-1}	2.5×10^{-1}	1.4×10^{-1}	9.0×10^{-2}	1.4×10^{-3}	1.3×10^{-13}	0.0
^{153}Gd	1.1×10^{-2}	3.2×10^{-7}	9.4×10^{-12}	0.0	0.0	0.0	0.0
^{154}Eu	2.0×10^{2}	1.3×10^{2}	8.6×10^{1}	5.5	2.7	8.0×10^{-8}	0.0
^{155}Eu	2.2×10^{-1}	4.8×10^{-1}	1.0×10^{-2}	1.8×10^{-4}	0.0	0.0	0.0
^{166m}Ho	4.0×10^{-5}	4.0×10^{-5}	3.9×10^{-5}	3.9×10^{-5}	3.8×10^{-5}	3.0×10^{-5}	2.2×10^{-5}
Subtotal	1.7×10^{4}	1.1×10^{4}	8.3×10^{3}	6.4×10^{3}	1.2×10^{3}	2.4	1.1×10^{-1}
Actinides and Daughters							
^{227}Ac, ^{223}Ra, ^{219}Rn, ^{215}Po, ^{211}Pb, ^{211}Bi, ^{207}Tl[(b)]	2.7×10^{-6}	4.3×10^{-6}	5.9×10^{-6}	7.6×10^{-6}	1.8×10^{-5}	7.5×10^{-5}	1.7×10^{-4}
^{288}Th, ^{244}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi[(b)]	7.9×10^{-3}	7.7×10^{-3}	7.1×10^{-3}	6.1×10^{-3}	3.3×10^{-3}	7.0×10^{-5}	5.7×10^{-7}
^{212}Po	8.4×10^{-4}	8.2×10^{-4}	7.6×10^{-4}	6.6×10^{-4}	3.5×10^{-4}	7.4×10^{-6}	6.1×10^{-8}
^{208}Tl	4.7×10^{-4}	4.6×10^{-4}	4.2×10^{-4}	3.7×10^{-4}	2.0×10^{-4}	4.2×10^{-6}	3.4×10^{-8}

TABLE 8.5.1. (contd)

Actinides and Daughters	Radioactivity, Ci Shutdown	10 Yr	20 Yr	30 Yr	100 Yr	500 Yr	1000 Yr
^{212}Po	8.4×10^{-4}	8.2×10^{-4}	7.6×10^{-4}	6.6×10^{-4}	3.5×10^{-4}	7.4×10^{-6}	6.1×10^{-8}
^{208}Ti	4.7×10^{-4}	4.6×10^{-4}	4.2×10^{-4}	3.7×10^{-4}	2.0×10^{-4}	4.2×10^{-6}	3.4×10^{-8}
^{230}Th	4.5×10^{-6}	7.8×10^{-6}	1.2×10^{-5}	1.7×10^{-5}	6.2×10^{-5}	5.0×10^{-4}	1.1×10^{-3}
^{231}Pa	9.5×10^{-7}	1.2×10^{-6}	1.4×10^{-6}	1.6×10^{-6}	3.0×10^{-6}	1.1×10^{-5}	2.1×10^{-5}
^{234}Pa	1.9×10^{-5}	1.9×10^{-5}	1.9×10^{-5}	1.9×10^{-5}	1.9×10^{-5}	1.9×10^{-5}	1.9×10^{-5}
^{232}U	1.3×10^{-3}	1.3×10^{-3}	1.2×10^{-3}	1.0×10^{-3}	5.3×10^{-4}	1.1×10^{-5}	9.2×10^{-8}
^{233}U	3.9×10^{-6}	5.0×10^{-6}	6.1×10^{-6}	7.6×10^{-6}	1.7×10^{-5}	1.1×10^{-4}	2.8×10^{-4}
^{234}U	3.3×10^{-2}	4.1×10^{-2}	4.9×10^{-2}	5.7×10^{-2}	9.2×10^{-2}	1.4×10^{-1}	1.4×10^{-1}
^{235}U, ^{231}Th(b)	1.9×10^{-3}	1.9×10^{-3}	1.9×10^{-3}	1.9×10^{-3}	1.9×10^{-3}	1.9×10^{-3}	2.0×10^{-3}
^{236}U	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.7×10^{-2}	1.7×10^{-2}
^{237}U	1.4×10^{-1}	8.8×10^{-2}	5.5×10^{-2}	3.0×10^{-2}	1.3×10^{-3}	2.8×10^{-6}	2.5×10^{-6}
^{238}U, ^{234}Th, ^{234m}Pa(b)	5.7×10^{-2}	5.7×10^{-2}	5.7×10^{-2}	5.7×10^{-2}	5.7×10^{-2}	5.7×10^{-2}	5.7×10^{-2}
^{237}Np, ^{233}Pa(b)	5.02×10^{-2}	5.1×10^{-2}	5.3×10^{-2}	5.6×10^{-2}	7.1×10^{-2}	1.4×10^{-1}	1.8×10^{-1}
^{230}Pu	2.7×10^{-3}	2.3×10^{-4}	2.0×10^{-5}	8.4×10^{-7}	0.0	0.0	0.0
^{238}Pu	3.0×10^{2}	2.8×10^{2}	2.6×10^{2}	2.4×10^{2}	1.4×10^{2}	6.5	1.8×10^{-1}
^{239}Pu	2.2×10^{1}	2.2×10^{1}	2.2×10^{1}	2.2×10^{1}	2.2×10^{1}	2.1×10^{1}	2.1×10^{1}
^{240}Pu	4.4×10^{1}	4.4×10^{1}	4.5×10^{1}	4.5×10^{1}	4.5×10^{1}	4.3×10^{1}	4.1×10^{1}
^{241}Pu	5.6×10^{3}	3.5×10^{2}	2.2×10^{3}	1.2×10^{3}	5.2×10^{1}	1.1×10^{-1}	1.0×10^{-1}
^{242}Pu	2.3×10^{-1}	2.3×10^{-1}	2.3×10^{-1}	2.3×10^{-1}	2.3×10^{-1}	2.3×10^{-1}	2.3×10^{-1}
^{241}Am	2.0×10^{2}	2.7×10^{2}	3.1×10^{2}	3.4×10^{2}	3.4×10^{2}	1.8×10^{2}	8.2×10^{1}
^{242m}Am, ^{242}Am(b)	5.4	5.2	5.0	4.5	3.5	5.6×10^{-1}	5.7×10^{-2}
^{243}Am, ^{239}Np(b)	5.6	5.7	5.7	5.7	5.6	5.4	5.2
^{242}Cm	1.5×10^{1}	2.1	2.0	1.9	1.4	2.3×10^{-2}	2.3×10^{-2}
^{243}Cm	4.3×10^{-1}	3.5×10^{-1}	2.8×10^{-1}	2.1×10^{-1}	4.9×10^{-2}	8.5×10^{-6}	1.7×10^{-10}
^{244}Cm	2.6×10^{2}	1.7×10^{2}	1.2×10^{2}	7.2×10^{1}	5.5	1.2×10^{-6}	0.0
^{245}Cm	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}	1.0×10^{-1}
^{246}Cm	2.0×10^{-2}	2.0×10^{-2}	2.0×10^{-2}	2.0×10^{-2}	2.0×10^{-2}	1.9×10^{-2}	1.8×10^{-2}
^{247}Cm, ^{243}Pu(b)	2.1×10^{-7}	2.1×10^{-7}	2.1×10^{-7}	2.1×10^{-7}	2.1×10^{-7}	2.1×10^{-7}	2.1×10^{-7}
^{248}Cm	4.5×10^{-7}	4.5×10^{-7}	4.5×10^{-7}	4.5×10^{-7}	4.5×10^{-7}	4.5×10^{-7}	4.5×10^{-7}
^{249}Cf	7.7×10^{-6}	7.7×10^{-6}	7.4×10^{-6}	7.2×10^{-6}	6.4×10^{-6}	2.9×10^{-6}	1.1×10^{-6}
^{250}Cf	1.0×10^{-5}	6.1×10^{-6}	3.6×10^{-6}	2.0×10^{-6}	5.2×10^{-8}	2.9×10^{-14}	0.0
^{251}Cf	2.1×10^{-7}	2.0×10^{-7}	2.0×10^{-7}	2.0×10^{-7}	1.9×10^{-7}	1.4×10^{-7}	9.5×10^{-8}
^{252}Cf	2.6×10^{-6}	1.9×10^{-6}	1.4×10^{-8}	1.0×10^{-9}	0.0	0.0	0.0
Subtotal	6.5×10^{3}	4.3×10^{3}	3.0×10^{3}	1.9×10^{3}	6.2×10^{2}	2.6×10^{2}	1.5×10^{2}
Total	2.4×10^{4}	1.5×10^{4}	1.1×10^{4}	8.3×10^{3}	7.4×10^{3}	2.6×10^{2}	1.5×10^{2}

a. Based on an accumulation of 2 kg spent fuel products per year over the 30-year lifetime of the plant. Activities listed reflect radiation levels after completion of predecommissioning chemical decontamination procedures. Year 2000 spent fuel used as reference reprocessing feed. Fuel is assumed to be reprocessed 1.5 years after reactor discharge.

b. Total activity (parent plus daughters) listed for decay chains with parent and short half-life daughter(s).

8.5.2.1 Decommissioning Plan and Schedule of Events

A description of the activities required to place a retired facility in passive safe storage is given in Section 8.2.2. An approximate schedule of events for placing the FRP in safe storage is presented in Figure 8.5.1.

The planning and preparatory activities are carried out during the final two years of plant operations. After facility shutdown activities are completed, the main process areas are chemically decontaminated. Used chemical decontamination solutions are processed through the facility intermediate-level liquid waste treatment system. Radioactive contamination outside the process cells is removed or fixed in place. The water in the fuel storage pool and SHLW canister storage pools is treated when necessary by deionization, filtration, and evaporation so that the water released to the environment has radioactivity levels below the limits specified in Reference 1. Residual contamination on the pool walls, floors, or equipment is fixed in place. All equipment in the facility except essential safety systems that will remain in operation during the continuing care period is deactivated. Areas containing significant radioactive contamination are isolated from the remainder of the facility. The ventilation system is deactivated and contaminated portions are isolated. Most of the radioactive contamination remaining in the facility during the continuing care period would be isolated in the main process cells and waste treatment cells.

Security during the continuing care period is provided by the exclusion area fence, electronic surveillance equipment, and locked or disabled exterior doors into the facility. Routine inspections and radiation and environmental surveys are conducted throughout the continuing care period.

An approximate schedule of events for dismantling the reference FRP at the end of the continuing care period is presented in Figure 8.5.2. Planning and preparation for the dismantlement takes place during the final two years of the continuing care period. The initial activity in the dismantlement operation is the renovation of effluent control systems and other facility equipment required to be in operation during the dismantlement. Contaminated equipment is then removed from the areas that were isolated for safe storage, and structural surfaces in these areas are mechanically decontaminated. Contaminated equipment and structures remaining in other portions of the facility are then removed. The ventilation system is the final portion of each section of the facility to be decontaminated. All contaminated materials are packaged and shipped offsite for disposal. At the discretion of the facility owner, the decontaminated structures may be demolished and the site restored approximately to its pre-facility condition.

8.5.2.2 Manpower Requirements

The estimated manpower requirements for safe storage, 30 years of continuing care, and deferred dismantlement of the reference FRP are summarized in Table 8.5.2. These manpower estimates are based on results from the NRC decommissioning study,[7] modified as necessary for the reference facility used in this study.

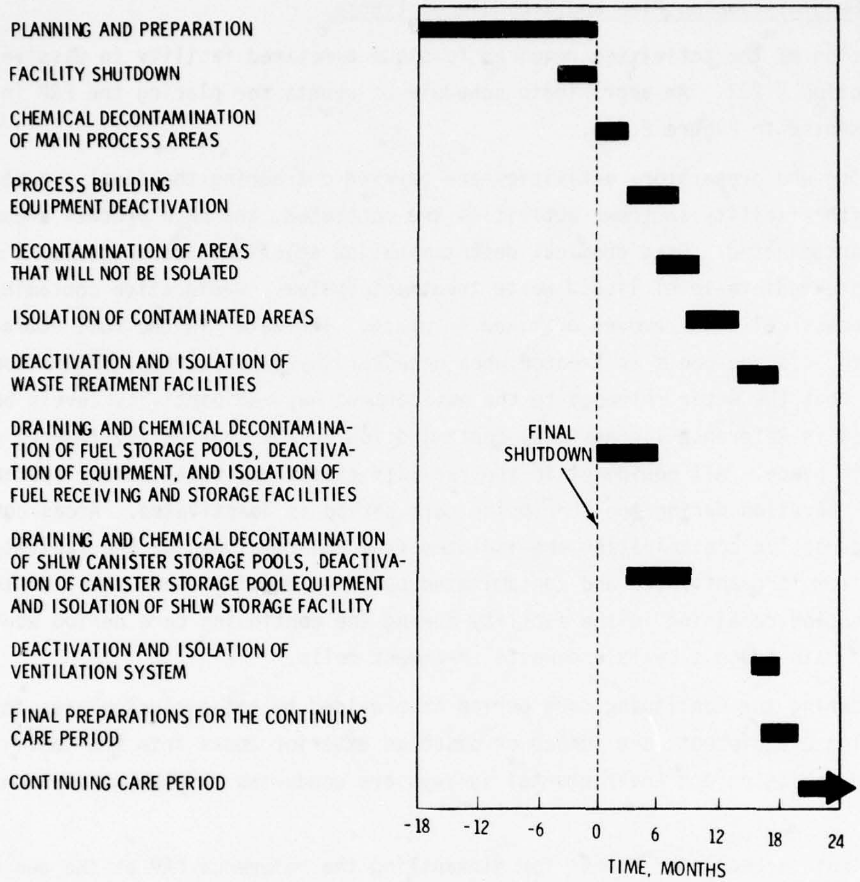


FIGURE 8.5.1. Approximate Schedule of Events for Placing the Fuel Reprocessing Plant in Safe Storage

8.5.2.3 Radioactive Wastes

The estimated volumes of radioactive waste generated during safe storage, continuing care, and dismantlement of the reference FRP are presented in Table 8.5.3. Most combustible and wet wastes generated during safe storage operations are assumed to be treated onsite with installed waste treatment equipment, i.e. converted to a noncombustible solid and stored in nonflammable containers in the facility until it is dismantled.

About 25 m^3 of combustible wastes would be generated at the end of the activities required to place the facility in safe storage, after the waste treatment facilities have been shut down. These wastes would be packaged and shipped offsite to a disposal or treatment facility. About 20 m^3 of additional combustible wastes are estimated to be generated during the 30-yr continuing period. These wastes are also assumed to be packaged and shipped offsite to a disposal or waste treatment facility as they are generated.

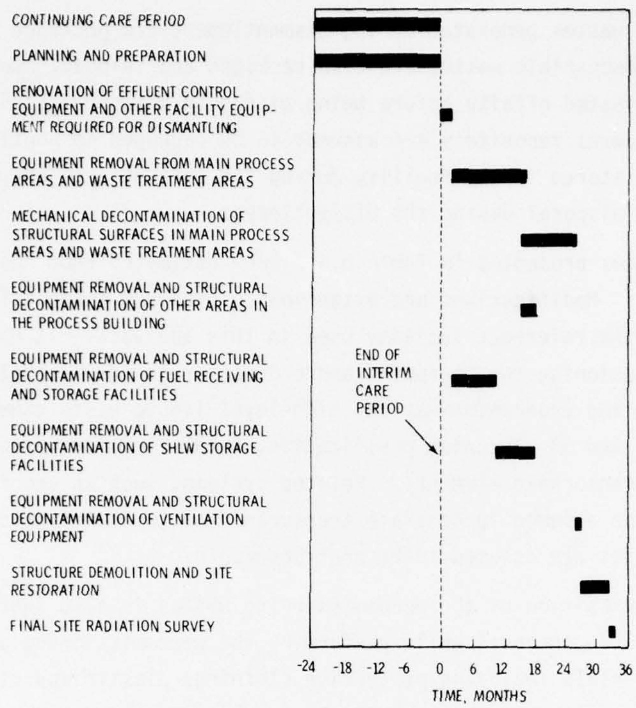


FIGURE 8.5.2. Approximate Schedule of Events for Deferred Dismantlement of the Fuel Reprocessing Facility

TABLE 8.5.2. Estimated Manpower Requirements for Safe Storage-Dismantlement of the Reference Fuel Reprocessing Plant

Job Description	Man-Years: Protective Storage	30 Year Interim Care	Deferred Dismantlement	Total
Facility operations and maintenance	25	12	40	80
Engineering and supervisory	9	18	35	60
Health physics and laboratory	5	--	10	15
Radiation protection	5	15	10	30
Skilled decommissioning	24	--	50	70
Management, clerical and security	24	15	60	100
Total	90	60	200	350

Noncombustible wastes generated during dismantlement are packaged without treatment and shipped offsite. Combustible wastes are also packaged and shipped. Combustible TRU wastes are assumed to be treated offsite before being placed in a Federal repository. All wastes destined for the Federal repository are assumed to be packaged in nonflammable containers. The packaged wastes stored in the facility during the safe storage operations would also be shipped offsite for disposal during the dismantlement.

The waste volumes presented in Table 8.5.3 were estimated from results of the NRC Decommissioning study.[7] Modifications and extensions of the NRC information were made to account for differences in the reference facility used in this analysis. It has been assumed that all wastes from decommissioning the equipment and process areas for 1) fuel shearing and dissolution; 2) feed preparation and codecontamination, high-level liquid waste concentration, interim storage and solidification; and 3) plutonium purification, solidification, packaging and storage are contaminated with transuranic elements. Related systems, such as ventilation equipment serving these areas, are also assumed to generate transuranic-contaminated decommissioning wastes. Other decommissioning wastes are assumed to be nontransuranic.

The general composition of the decommissioning wastes is also indicated in Table 8.5.3. Combustible wastes are generated while performing the decommissioning operations. They consist of a variety of materials including protective clothing, plastic and cloth coverings, and wood. The waste is similar to the combustible trash generated during normal plant operations and maintenance activities. Wet wastes include resins, water filters and sludges from storage pool water treatment systems, evaporator bottom, and concentrated chemical decontamination solutions. Noncombustible wastes include stainless steel and some titanium process equipment, aluminum fuel and high-level waste canister storage racks, stainless steel process cell and storage pool liners, glove boxes, concrete rubble and HEPA filters. The estimated radioactive contamination levels on the waste are also indicated in the table.

8.5.2.4 Routine Radioactive Effluents. Routine radioactive effluents during decommissioning are generally atmospheric releases of gases and particulates. A variety of decommissioning operations can cause radioactive materials to become airborne inside the facility. Decommissioning operations that cause radioactive materials to become airborne include agitating and spraying of chemical decontamination solutions in process vessels; high and low pressure decontamination solution spraying; jackhammering, drilling and rock splitting, and explosive removal of concrete; and plasma torch cutting of stainless steel equipment. Procedures are followed to reduce the amount of these airvorne materials that reach the ventilation system. Mist eliminators are installed to remove airborne decontamination solutions. Water sprays may be used to reduce dust generation during drilling, jackhammering, rock splitting or explosive blasting of concrete. Portable fume hoods may be positioned over equipment being cut with plasma torches. Nevertheless, a small fraction of the airborne radioactive materials can be expected to be released to the atmosphere through the facility stack after passing through the facility APS and off-gas treatment system.

TABLE 8.5.3. Estimated Waste Volumes for Safe Storage-Deferred Dismantlement of the Reference Fuel Reprocessing Plant

Waste Type	Decommissioning Phase	Volume m^3(a)	Waste Description	Radionuclide Content Fraction(b) F.P.	Radionuclide Content Fraction(b) Actinides	Waste Disposition	Time of Shipment
TRU							
Compactible waste and combustible trash	Safe storage	170	HEPA filters	.14	.14	Packaged, shipped for treatment	Dismantlement
	Safe storage	30	HEPA filters	0	.20	Packaged, shipped for treatment	Dismantlement
	Dismantlement	170	HEPA filters	.01	.01	Packaged, shipped for treatment	Dismantlement
	Dismantlement	30	HEPA filters	0	.02	Packaged, shipped for treatment	Dismantlement
	Safe storage	100	Combustible trash	2×10^{-4}	2×10^{-4}	Treated, packaged, shipped for disposal	Dismantlement
	Dismantlement	130	Combustible trash	4×10^{-4}	4×10^{-4}	Packaged, shipped for treatment	Dismantlement
Noncompactible, non-combustible waste	Dismantlement	180	Equipment and structural materials	.33	.30	Packaged, shipped for disposal	Dismantlement
	Dismantlement	780	Equipment and structural materials	.16	.13	Packaged, shipped for disposal	Dismantlement
	Dismantlement	1,200	Equipment and structural materials	.10	.10	Packaged, shipped for disposal	Dismantlement
Wet wastes and liquids	Safe storage	230	Noncombustible wet wastes	.10	.10	Treated, packaged, shipped for disposal	Dismantlement
NON-TRU							
Compactible waste and combustible trash	Safe storage	25	HEPA filters	.02	0	Packaged, shipped for disposal	Dismantlement
	Dismantlement	25	HEPA filters	.002	0	Packaged, shipped for disposal	Dismantlement
	Safe storage	60	Combustible trash	1×10^{-4}	0	Treated, packaged, shipped for disposal	Dismantlement
	Safe storage	15	Combustible trash	5×10^{-5}(c)	0	Packaged, shipped for treatment/disposal	Safe storage
	Continuing care	10	Combustible trash	4×10^{-5}(c)	0	Packaged, shipped for treatment/disposal	Continuing care
	Dismantlement	80	Combustible trash	1×10^{-5}	0	Packaged, shipped for treatment	Dismantlement
Noncompactible, non-combustible waste	Dismantlement	180	Equipment and structural materials	.02	0	Packaged, shipped for disposal	Dismantlement
	Dismantlement	6,000	Equipment and structural materials	.02	0	Packaged, shipped for disposal	Dismantlement
Wet wastes and liquids	Safe storage	140	Combustible wet wastes	.07	0	Treated, packaged, shipped for disposal	Dismantlement
	Safe storage	60	Noncombustible wet wastes	.03	0	Treated, packaged, shipped for disposal	Dismantlement

a. All volumes are raw waste before treatment or packaging.
b. Fraction of total activity shown in 30 year column of Table 8.5.1 in that waste stream, unless otherwise indicated.
c. Reference Table 8.5.1 - shutdown column.

In the NRC study it was estimated that routine effluents during the protective storage operations would be less than 2 mCi of mixed fission products.[7] This corresponds to a release of material with a radioactivity level 2×10^{-7} times the shutdown column of the fission product portion of Table 8.5.1. Plutonium would be expected to be a very small fraction of the total release, since portions of the facility containing significant amounts of plutonium have an additional stage of HEPA filters to remove contamination from the ventilation exhaust. The actinide release is estimated to be equivalent to 10^{-10} times the actinide portion of the shutdown column of Table 8.3.16. No measureable routine releases are expected during the continuing care period. Routine releases during the dismantling operations are expected to be about 1 mCi or less of mixed fission products. This corresponds to a release of material with a radioactivity content 2×10^{-7} times the 30-year column of the fission product estimates in Table 8.5.1. Actinide releases are estimated to be equivalent to 10^{-10} times the actinide portion of the table.

Radioactive materials will also be released to the environment during the safe storage operations when the fuel storage pool and SHLW canister storage pools are drained and when chemical decontamination solutions are concentrated. The pool water will be treated by filtration, deionization and evaporation before it is released. Most of the released radioactivity is expected to be released from evaporator overheads to the atmosphere through the facility stacks. Some evaporator overheads may also be condensed and released to local water bodies. Consistent with ALARA principles, all effluents will be well below the requirements of 10 CFR 20 for release of radioactive materials to the environment. A total of about 12,000 m^3 of water slightly contaminated with a mixture of fission products is expected to be released as condensate from draining the fuel storage pools. About 6,000 m^3 of water also slightly contaminated with a mixture of fission products will be released as condensate when the SHLW canister storage pools are drained. This release is estimated to have a radioactivity content equivalent to 5×10^{-5} times column 8 of reference Table 3.3.4. About 1,000 m^3 of condensate water is expected to be released from concentrating chemical decontamination solutions. Contamination in this water will be a mixture of fission products and actinides. This release is estimated to have a radioactivity content equivalent to 10^{-7} times the shutdown column of Table 8.5.1.

8.5.2.5 Cost Estimate

The estimated costs for the safe storage, continuing care, and dismantling operations of the FRP are presented in Table 8.5.4. These costs are based on information from a recent decommissioning study,[7] modified as necessary for the reference facility and basic assumptions of this study. The procedures outlined in this study assume that the plant had a relatively normal operating history and that decommissioning operations proceed with a minimum of difficulties. A 25% contingency was added to all cost estimates to account for unforeseen circumstances. Costs are presented in thousands of mid-1976 dollars. Continuing care and deferred dismantlement costs were not discounted.

TABLE 8.5.4. Estimated Costs for Safe Storage-Deferred Dismantlement of the Fuel Reprocessing Plant, $1,000

Cost Element	Safe Storage	30 Year Continuing Care	Deferred Dismantling	Total
Labor	2,300	1,500	5,100	9,000
Equipment and supplies	1,000	800	4,000	6,000
Utilities and services	700	800	1,500	3,000
Demolition and site restoration	NA	NA	3,000	3,000
Owner costs	1,000	800	4,000	7,000
Contingency	1,000	800	4,000	7,000
Total	6,000	5,000	22,000	36,000

Labor costs indicated in Table 8.3.19 include direct wages, fringe benefits, and other direct employer expenses. The allowance for utilities and services is to cover the costs of electricity, boiler fuels, water and water treatment, sewage treatment or disposal, and other utilities and services required during the decommissioning operation. Equipment costs include development and design costs for specialized decommissioning equipment, allowances for other equipment that must be purchased, as well as expendable equipment and supplies and waste packaging. Owner costs includes expenses incurred directly by the owner during planning and performance of the decommissioning operations. These costs include overheads such as engineering, operating and accounting expenses, insurance, taxes, permits, licenses, and the cost of making reports and applications to regulatory agencies.

8.5.2.6 Nonradiological Impacts

The nonradiological impacts of decommissioning the reference FRP by the safe storage-deferred dismantlement mode are discussed below. Where it has not been possible to quantify the impacts, they have been related to similar impacts during construction or operation of the facility. In general, the decommissioning impacts are expected to be less than similar impacts during construction and operation.

Land Use. The 24 km^2 (6000 acre) site is maintained throughout the safe storage and 30-year continuing care period. When dismantlement is completed the site may be released for alternate uses at the discretion of the facility owner.

Water Use. Requirements for water during decommissioning are reduced substantially from requirements during plant operations. Sanitary water usage is decreased since the decommissioning work force is less than the operating work force. The water used to cool the fuel storage and SHLW canister pools during plant operation is no longer required. Water used in preparing aqueous chemicals for use in the reprocessing operations is not required after the chemical decontamination phase of the decommissioning is completed.

Equipment and Materials. Some materials and equipment used for decommissioning become radioactively contaminated and are removed from further use following the decommissioning operations. These include steel radioactive waste disposal containers; cloth, paper, plastic, and wood used for personnel protection and contamination control; and specialized decommissioning equipment such as cutting torches, temporary shielding, hand tools, rock splitters and so forth. The approximate amounts of material used for this decommissioning mode are as follows: steel shipping containers, 1600 MT; paper, wood, plastic, 130 MT; and equipment (mostly steel), 500 MT.

Energy. Electricity usage during decommissioning should be less than during plant operations. The primary use of electricity during decommissioning is for operating the ventilation system (about 25 MW-hr for safe storage-dismantlement). Boiler fuel will be burned for space heating and for operating the evaporators. Diesel fuel and gasoline will be burned in plant vehicles and equipment used during decommissioning. Heavy equipment used during the demolition and site restoration phase of dismantlement will be the primary consumer of vehicle fuels.

Air Quality Effects. During demolition and site restoration, quantities of fugitive dust will be generated from building demolition, loading of concrete rubble, grading and backfilling operations. Vehicle effluents will also be produced. The effects will be transitory and will be confined to the immediate vicinity of the site. Areas to be graded or excavated will be wet down to minimize dust generation.

Noise Impacts. Noise caused by decommissioning will vary with day-to-day work schedules, weather conditions, and other factors. Noise during the safe storage operations and the continuing care period are expected to be less than noise during present operations. During dismantling, the noisiest phase of work will be the demolition and site restoration phase, which would take about 6 months. Noise levels during these operations are expected to be comparable to the noise levels during facility construction.

Effects on Nearby Communities. The major impact on local communities would be associated with staff reduction during the decommissioning. The plant work force would be reduced from about 400 people during normal operations to about 60 during the protective storage operations and to the equivalent of about two persons during the 30-year continuing care period. The work force would increase again to about 75 people during dismantlement and gradually be reduced to zero when dismantlement is complete.

8.5.3 Hardened Safe Storage of the Reference Fuel Reprocessing Plant

To place the reference fuel reprocessing plant in hardened safe storage, all residual radioactivity in the facility is confined to portions of the plant that are separated from the remainder of the facility and the environment by permanent physical barriers. For this study, it is assumed that the facility is maintained in this condition for an extended period of time (perhaps as long as 1000 years). Because the facility contains significant amounts of long half-life radionuclides, activities would be required at the end of the continuing

care period to finally decommission the facility and terminate the license. These activities would include removal of all residual radioactive materials in the plant above levels designated by regulatory agencies for public access to the facility. Significant refurbishing of the facility may be required to permit the safe performance of these activities.

8.5.3.1 Decommissioning Plan and Schedule of Events

A description of the general types of activities required to place a retired nuclear facility in hardened safe storage is given in Section 8.2.2. An approximate schedule of events for hardened safe storage of the reference fuel reprocessing facility is presented in Figure 8.5.3. Planning and preparation takes place during the final one to two years of facility operation. After facility shutdown activities are complete, the main process areas are chemically decontaminated. Used decontamination solutions are processed through the facility intermediate-level liquid waste treatment system.

After the equipment in the main process building is deactivated, all materials with radioactive contamination outside the main process cells and high-level liquid waste storage and treatment areas is removed and placed in one of the cells. The main process cells and high-level liquid waste storage and treatment areas are separated from the remainder of the facility by the installation of permanent physical barriers. Any radioactive contamination remaining outside the isolation structure is removed, packaged, and shipped offsite.

The water in the fuel storage pool and SHLW canister storage pools is deionized, filtered, evaporated and released to the environment under the provisions of 10 CFR 20. The pool walls and floor are chemically decontaminated. All contaminated water treatment, pumping and cooling equipment, and shipping cask handling equipment is removed and placed in the storage pools. The storage pools then are separated from the remainder of the facility by permanent physical barriers.

The ventilation system is the final portion of the facility to be deactivated. All contaminated equipment, filters, and ductwork are packaged and shipped offsite for disposal. Most exterior doors into the facility are disabled. The exclusion area fence is maintained, and the remainder of the site could be released for nonnuclear use. Activities during the continuing care period include environmental monitoring and inspection of the isolation structures. The facility is protected from unauthorized entry by the exclusion fence and locked or disabled exterior doors.

8.5.3.2 Manpower Requirements

The estimated manpower requirements for hardened safe storage of the fuel reprocessing plant are presented in Table 8.5.5. The manpower requirements for one year of continuing care are also given in the table. However, continuing care activities are assumed to be carried out for periods up to 1000 years. Estimated manpower for finally decommissioning the facility at the end of the continuing care period is also included in the table.

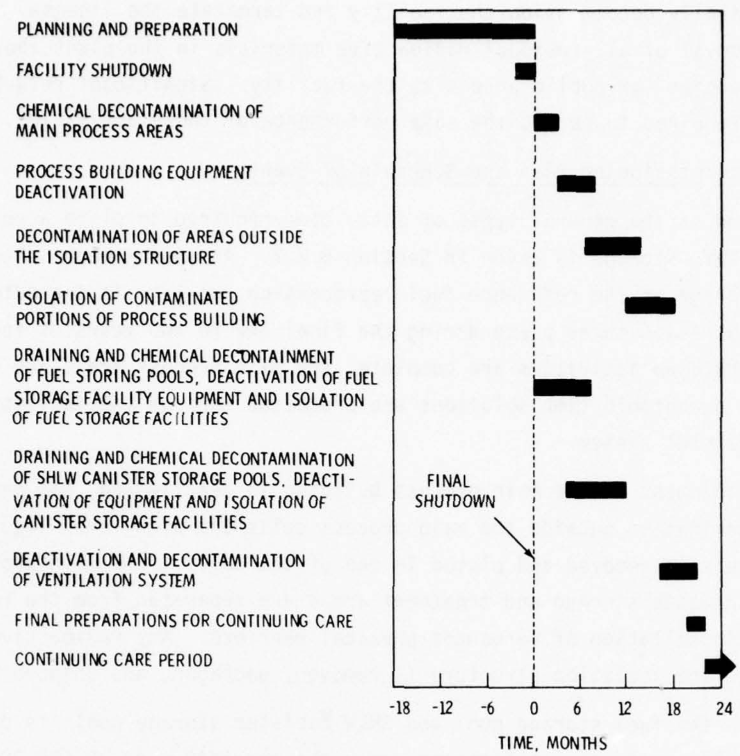


FIGURE 8.5.3. Approximate Schedule of Events for Hardened Safe Storage of the Reference Fuel Reprocessing Facility

TABLE 8.5.5. Estimated Manpower Requirements for Hardened Safe Storage of the Reference Fuel Reprocessing Plant

Job Description	Man-Years: Safe Storage	Man-Years: Continuing Care per year[a]	Man-Years: Final Decommissioning
Facility operation and maintenance	30		20
Engineering and supervisory	25	0.1	30
Health physics and laboratory	6		10
Radiation protection	6	0.1	10
Skilled and semiskilled decommissioning	35		40
Management, clerical and security	45	0.1	50
Total (Rounded)	150	0.3	160

a. Continuing care activities are carried out indefinitely. Manpower requirements per year are shown.

8.5.3.3 Radioactive Wastes

Most of the radioactivity in the FRP at shutdown remains in the facility. However, there are some wastes generated during the safe storage operations that will require offsite disposal. These wastes are primarily those wastes generated in the final cleanup operations after the isolation structures have been closed and combustible wastes have been generated after the onsite combustible waste treatment facility has been decommissioned.

Combustible and wet wastes generated during the first phases of the safe storage operation are assumed to be converted to a nonflammable solid in the installed waste treatment facilities and placed in the isolation structure in nonflammable containers. The estimated volumes of waste requiring offsite disposal during the safe storage operation are summarized in Table 8.5.6. Less than 1 m^3/yr of radioactive wastes are estimated to be generated during the continuing care period.

TABLE 8.5.6. Estimated Volumes of Radioactive Waste Shipped Offsite During Hardened Safe Storage of the Reference Fuel Reprocessing Plant

Waste Type	Volume, m^3 (a)	Waste Description	Radionuclide Content Fraction(b) F.P.	Radionuclide Content Fraction(b) Actinides	Waste Disposition
TRU					
Compactible waste and combustible trash	170	HEPA filters	.08	.08	Packaged, shipped for treatment
	30	HEPA filters	0.0	.15	Packaged, shipped for treatment
Non-TRU					
Compactible waste and combustible trash	25	HEPA filters	.01	0	Packaged, shipped for disposal
	30	Combustible trash	5×10^{-5}	0	Packaged, shipped for disposal
Noncompactible, non-combustible waste	60	Equipment and structural materials	1×10^{-4}	0	Packaged, shipped for disposal

a. Raw waste volumes are given for all wastes.
b. Fraction of activity shown in shutdown column of Table 8.5.1 in that waste stream.

Combustible wastes generated during the latter stages of the decommissioning operation are typical of combustible wastes generated during plant operations and maintenance activities. They are composed of protective clothing, plastic and cloth coverings, and wood.

Noncombustible wastes shipped offsite during the safe storage operations are primarily produced from the decontamination of the ventilation system. They include HEPA filters, stainless steel ductwork and equipment, and concrete rubble.

8.5.3.4 Routine Radioactive Effluents

Routine radioactive effluents during the hardened safe storage operations are expected to be similar to the effluents released during the passive safe storage phase of safe storage-deferred dismantlement. These effluents were described in the Routine Radioactive Effluents

under Section 8.3.2.4. Less than 1 mCi of mixed fission products is estimated to be released through the facility ventilation system as a result of the hardened safe storage operations. This represents 4×10^{-8} times the shutdown column of Table 8.5.1. Actinide releases are expected to be about a factor of 1000 lower or 4.0×10^{-11} times the shutdown column of Table 8.5.1. Effluents from draining the fuel storage pool and the SHLW canister storage pools and from concentrating chemical decontamination solutions will be the same for hardened safe storage as for passive safe storage.

8.5.3.5 Cost Estimate

The estimated costs for hardened safe storage of the reference FRP are presented in Table 8.5.7. The assumptions, definitions, and limitations associated with this cost estimate are similar to those presented for safe storage-deferred dismantlement. The estimated annual cost of continuing care and the final decommissioning costs are also indicated in the table. Most of the cost of continuing care is associated with owner costs such as taxes and insurance. Continuing care activities at an entombed FRP are assumed to be carried out indefinitely.

TABLE 8.5.7. Estimated Costs for Hardened Safe Storage of the Reference Fuel Reprocessing Plant

Cost Element	Cost, 1000s of Mid-1976 Dollars: Safe Storage	Continuing Care Per Year	Final Decommissioning
Labor	3,800	8	4,000
Equipment and supplies	2,000	5	1,000
Utilities and services	1,000	2	500
Owner costs	1,500	15	1,400
Contingency	1,500	7	1,400
Total	10,000	40	8,000

8.5.3.6 Nonradiological Impacts

The nonradiological impacts of hardened safe storage of the reference FRP are discussed briefly below. Where it has not been possible to quantify the impacts, they have been related to similar impacts during construction or operation of the facility. In general, the decommissioning impacts are expected to be less than similar impacts during construction or normal operation of the facility.

Land Use. The 0.24 km^2 (60 acre) exclusion area of the facility site is occupied indefinitely by the hardened facility. The remaining 24 km^2 of the site could be released for alternate uses at the discretion of the facility owner. Activities near the exclusion area are restricted to those that will not jeopardize the integrity of the isolation structure. Land would also be required in a commercial burial ground to dispose of the radioactive decommissioning wastes.

Water Use. Requirements for water during decommissioning are reduced substantially from requirements during plant operations. Sanitary water usage is decreased since the decommissioning work force is less than the operating work force. The water used to cool the fuel storage and SHLW canister storage pools during plant operation is no longer required. Water used in preparing aqueous chemicals for use in the reprocessing operations is also not required after the chemical decontamination phase of the decommissioning is completed.

Equipment and Materials. Some materials and equipment used for decommissioning become radioactively contaminated and are removed from further use following the decommissioning operations. These include steel radioactive waste disposal containers; cloth, paper, plastic, and wood used for personnel protection and contamination control; and specialized decommissioning equipment such as cutting torches, temporary shielding, hand tools, rock splitters and so forth. The approximate amounts of material used for this decommissioning mode is as follows: steel shipping containers, 10 MT; paper, wood, plastic, 50 MT; and equipment (mostly steel), 100 MT.

Energy. Electricity usage during decommissioning should be less than during plant operations. The primary use of electricity during decommissioning is to operate the ventilation system (about 10 MW-hr for entombment). Diesel fuel and gasoline will be burned in plant vehicles and equipment used during decommissioning. Fossil fuels will be consumed for space heating and evaporator operation. The amount of vehicle and other fossil fuels consumed during decommissioning is expected to be less than the amounts consumed during plant operations.

Air Quality Effects. The hardened safe storage operations are not expected to noticeably affect air quality near the site. The primary source of effluents will be from vehicular traffic. Vehicular traffic during the decommissioning is expected to be less than the traffic during plant operations.

Noise Impacts. Noise caused by decommissioning will vary with day-to-day work schedules, weather conditions, and other factors. The primary source of noise during the hardened safe storage operations will be vehicular traffic, which is expected to be less than during plant operations.

Effects on Nearby Communities. The major impact on local communities will be associated with changes in the work force at the plant. The work force will be reduced from about 400 people during normal operations to about 100 people during the decommissioning operations to the equivalent of less than one person during the perpetual interim care period.

Institutional Effects. The continuing care activities at the facility must continue for at least several hundred years and perhaps longer. Institutions for assuring that the facility remains in a condition that poses minimum risk to the public and the environment for these long time periods do not exist.

8.5.4 Selection of a Decommissioning Alternative for the Reference Fuel Reprocessing Plant

Safe storage-deferred dismantlement has been selected as the reference alternative for decommissioning the fuel reprocessing plant. The facility is expected to contain significant amounts of transuranic contamination at the end of its operating lifetime. Hardened safe storage of the facility would require surveillance activities at the site to be carried out for long time periods to assure the continued protection of the public from the long-lived transuranic elements. Many uncertainties surround the surveillance and maintenance of the structure for very long time periods. Safe storage followed by dismantlement removes all the long-lived radioactivity to a deep geologic disposal site, releasing the FRP site for other uses.

8.6 DECOMMISSIONING OF A MIXED OXIDE FUEL FABRICATION PLANT

8.6 DECOMMISSIONING OF A MIXED OXIDE FUEL FABRICATION PLANT

Two decommissioning modes are considered for the reference mixed oxide fuel fabrication plant (MOX FFP). These include 1) immediate dismantlement and 2) hardened safe storage with an extended continuing care period. The absence of large amounts of shorter half-life radioactive contamination in the facility provides little incentive to consider deferred dismantlement. Hardened safe storage is considered for comparison, even though it may not be a viable alternative for a MOX FFP. Activities will be required at the end of the continuing care period to finally decommission the facility and terminate the license.

The analysis presented here assumes that facility shutdown activities are complete when decommissioning begins. Activities assumed to have been carried out to shut down the facility include: removal of all product and feed materials; processing, packaging and removal of all radioactive wastes generated during facility operations or shutdown activities; completion of preparations for closeout of special nuclear material accountability requirements; and removal from the site of hazardous chemicals and flammable materials not required for the decommissioning operations.

8.6.1 Reference Mixed-Oxide Fuel Fabrication Plant Description

The essential features of the reference 400-MTHM/yr mixed oxide fuel fabrication plant are presented in Section 3.2. To estimate the costs and identify potential environmental effects of decommissioning a MOX FFP it was necessary to make some assumptions concerning portions of the facility not discussed in Section 3.2.4. These assumptions are listed below.

- All equipment in the plutonium storage, plutonium rework, and pellet production and loading portions of the fuel fabrication process is operated and maintained remotely.
- Radioactive liquids and wet wastes are immobilized in an installed cementation (or bitumenization) facility before being shipped offsite for disposal.
- An incinerator is installed in the facility for treating combustible wastes. Incinerator ash is processed through the immobilization system before being shipped offsite for disposal.

To identify the potential environmental effects of decommissioning the reference MOX FFP, the amount of radioactive material remaining in the plant at the end of its operating lifetime was estimated. This estimate is presented in Table 8.6.1. The table also shows the decay of the residual radioactivity following shutdown. The table is based on the assumption that 1 kg of plutonium (with 22 kg of uranium) is held up in the plant each year over an operating lifetime of 30 years. This results in a plutonium inventory of about 28 kg in the plant at shutdown. The feed to the plant is assumed to be characteristic of the plutonium to be recycled in the year 2000. The feedstock for the plant is assumed to have been reprocessed one year before being used in the MOX FFP.

8.6.2 Dismantlement of the Reference Mixed Oxide Fuel Fabrication Plant

In the dismantlement mode, all radioactive material is removed from the MOX FFP to an approved disposal site. At the discretion of the facility owner, all buildings that contained radioactive materials during plant operation may be demolished and removed at the end of the dismantling operation and the site restored to approximately its prefacility condition.

8.6.2.1 Decommissioning Plan and Schedule of Events

A description of the activities required to dismantle a retired facility was given in Section 8.2.1. An approximate schedule of events for dismantlement of the MOX FFP is presented in Figure 8.6.1. The planning and preparatory activities are carried out during the final two years of facility operations.

TABLE 8.6.1. Estimated Inventory of Radioactive Materials in Reference Mixed Oxide Fuel Fabrication Plant at Shutdown and at Various Times Following Shutdown[a]

Nuclide	Radioactivity, Ci Shutdown	10 Yr	50 Yr	100 Yr	500 Yr	1000 Yr
^{227}Ac, ^{227}Th, ^{223}Ra, ^{219}Rn, ^{215}Po, ^{211}Pb, ^{211}Bi, ^{207}Tl[b]	7.4×10^{-6}	1.6×10^{-5}	6.8×10^{-5}	1.5×10^{-4}	8.6×10^{-4}	1.8×10^{-3}
^{228}Th, ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi[b]	1.7×10^{-1}	1.7×10^{-1}	1.2×10^{-1}	7.4×10^{-2}	1.6×10^{-3}	1.3×10^{-5}
^{212}Po	1.8×10^{-2}	1.8×10^{-2}	1.3×10^{-2}	7.9×10^{-3}	1.7×10^{-4}	1.4×10^{-6}
^{208}Tl	1.0×10^{-2}	1.0×10^{-2}	7.2×10^{-3}	4.4×10^{-3}	9.4×10^{-5}	7.7×10^{-7}
^{226}Ra, ^{222}Rn[c], ^{218}Po, ^{214}Pb, ^{214}Bi, ^{214}Po[b]	1.9×10^{-6}	5.1×10^{-6}	4.9×10^{-5}	2.2×10^{-4}	9.0×10^{-3}	4.1×10^{-2}
^{230}Th	8.4×10^{-5}	1.7×10^{-4}	7.5×10^{-4}	1.9×10^{-3}	1.8×10^{-2}	4.0×10^{-2}
^{231}Pa	3.5×10^{-6}	5.7×10^{-6}	1.4×10^{-5}	2.5×10^{-5}	1.1×10^{-4}	2.3×10^{-4}
^{234}Pa	2.2×10^{-4}	2.2×10^{-4}	2.2×10^{-4}	2.2×10^{-4}	2.2×10^{-4}	2.2×10^{-4}
^{232}U	2.8×10^{-2}	2.8×10^{-2}	1.9×10^{-2}	1.2×10^{-2}	2.5×10^{-4}	2.1×10^{-6}
^{233}U	9.3×10^{-7}	2.7×10^{-6}	2.4×10^{-5}	9.1×10^{-5}	1.9×10^{-2}	6.3×10^{-3}
^{234}U	8.0×10^{-1}	1.1	2.2	3.1	5.0	5.1
^{235}U, ^{231}Th[b]	2.0×10^{-2}	2.0×10^{-2}	2.0×10^{-2}	2.0×10^{-2}	2.1×10^{-2}	2.2×10^{-2}
^{236}U	8.4×10^{-4}	1.4×10^{-3}	3.4×10^{-3}	6.0×10^{-3}	2.6×10^{-2}	5.1×10^{-2}
^{237}U	5.5	3.4	5.3	5.1	3.7	0.0
^{238}U, ^{234}Th, ^{234M}Pa[b]	6.7×10^{-1}	6.7×10^{-1}	6.7×10^{-1}	6.7×10^{-1}	6.7×10^{-1}	6.7×10^{-1}
^{237}Np, ^{233}Pa[b]	2.6×10^{-2}	5.4×10^{-2}	2.1×10^{-1}	4.2×10^{-1}	1.6	2.4
^{236}Pu	8.7×10^{-2}	7.7×10^{-3}	4.6×10^{-7}	0.0	0.0	0.0
^{238}Pu	1.2×10^{4}	1.1×10^{4}	8.1×10^{3}	5.5×10^{3}	2.5×10^{2}	5.0
^{239}Pu	8.8×10^{2}	8.8×10^{2}	8.8×10^{2}	8.8×10^{2}	8.7×10^{2}	8.6×10^{2}
^{240}Pu	1.8×10^{3}	1.8×10^{3}	1.8×10^{3}	1.8×10^{3}	1.7×10^{3}	1.6×10^{3}
^{241}Pu	2.2×10^{5}	1.4×10^{5}	2.1×10^{4}	2.0×10^{3}	1.5×10^{-5}	0.0
^{241}Am	7.0×10^{3}	9.7×10^{3}	1.3×10^{4}	1.3×10^{4}	6.7×10^{3}	3.0×10^{3}
Total	2.4×10^{5}	1.6×10^{5}	4.5×10^{4}	2.3×10^{4}	9.5×10^{3}	5.5×10^{3}

a. Based on an accumulation of 1 kg Pu (with 22 kg U) per year over the 30 yr lifetime of the plant. MOX plant feed for year 2000 used as basis.
b. Total activity (parent plus daughters) listed for decay chains with parent and short half-life daughter(s).
c. ^{222}Rn and daughters present only after safe storage.

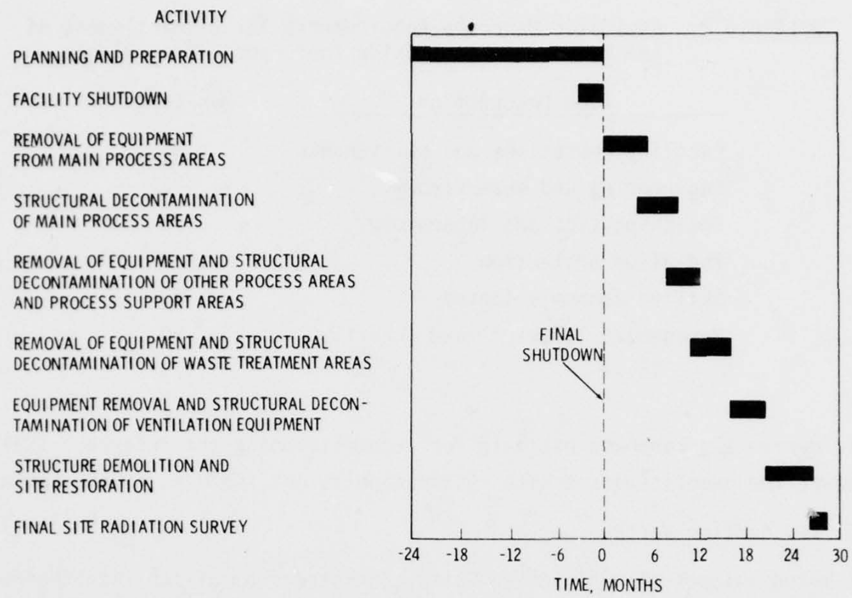


FIGURE 8.6.1. Approximate Schedule of Events for Dismantling the Reference Mixed Oxide Fuel Fabrication Plant

After facility shutdown activities have been completed, contaminated equipment is removed from the main process areas and the structural surfaces are decontaminated. The main process areas include equipment for mixing and blending the plutonium and uranium oxide; slugging, pressing, sintering, sizing, and grinding the fuel pellets; and recovery of clean and dirty scrap. Equipment and structures in these areas are expected to have the highest radioactive contamination levels in the plant.

Following decontamination of the main process areas, contaminated equipment is removed from the remaining process areas and process support areas, and structural surfaces in these areas are decontaminated. The remaining process areas include facilities for rod loading, welding, inspection, repair and disassembly. Process support areas include facilities for plutonium and uranium receiving, container storage, unloading and bulk storage of plutonium and uranium, and completed fuel rod storage and loadout. Radioactive contamination levels in these areas are expected to be relatively low.

These operations are followed by removal of equipment and structural decontamination of the waste treatment facilities installed in the plant. The final decommissioning operation involving radioactive material is the removal of contaminated equipment and structures from the ventilation system. The decontaminated structures may then be demolished and the site restored to approximately its pre-facility condition. A final radiation survey ensures that all potentially hazardous amounts of radioactivity have been removed from the site.

8.6.2.2 Manpower Requirements

The estimated manpower requirements for dismantling the MOX FFP are presented in Table 8.6.2. These manpower estimates are based on assumptions for staffing requirements consistent with

TABLE 8.6.2. Estimated Manpower Requirements for Dismantlement of the Reference Mixed-Oxide Fuel Fabrication Plant

Job Description	Man-Years
Facility operations and maintenance	30
Engineering and supervisory	25
Health physics and laboratory	12
Radiation protection	8
Skilled decommissioning	45
Management, clerical and security	50
Total	170

those used to derive the manpower estimate for decommissioning the reference fuel reprocessing plant. Manpower for demolition and site restoration is not included in the table.

8.6.2.3 Radioactive Wastes

The estimated volumes of radioactive waste generated from dismantling the reference MOX FFP are presented in Table 8.6.3. Estimates of radioactive contamination levels in the waste are also presented in the table.

All wet wastes and most combustible wastes are assumed to be converted to a noncombustible solid with onsite waste treatment equipment. All noncombustible waste and treated combustible and wet wastes are packaged in noncombustible containers and shipped to a Federal repository for disposal. About 20 m^3 of combustible waste are expected to be generated after the onsite waste treatment facilities have been shut down. These wastes are assumed to be packaged and shipped offsite for treatment and eventual disposal at the Federal repository.

The waste volumes presented are based on the information used to derive the estimated MOX FFP dismantlement wastes presented in Reference 3. Some adjustments were made in the information presented to assure consistency with the general assumptions used in this study. Waste contamination levels are based on values for wastes of similar composition generated during normal plant operation.

The general composition of the decommissioning wastes is given in the table. Combustible wastes are generated while performing the decommissioning operations. They consist of a variety of materials including protective clothing, plastic and cloth coverings, rags, paper, and wood. This waste is similar to the combustible trash generated during normal plant operations and maintenance activities. Wet wastes include concentrated TRU liquids, slurries, sludges, and spent resins. Noncombustible wastes are composed primarily of stainless and carbon steel process equipment, stainless steel process area liners and ventilation ductwork, glove boxes, concrete rubble, and HEPA filters.

8.6.2.4 Routine Radioactive Effluents

Routine radioactive effluents during dismantlement are generally atmospheric releases of gases and particles. A variety of dismantling operations can cause radioactive materials to become airborne inside the facility. These include wiping or spraying decontamination solutions;

TABLE 8.6.3. Estimated Waste Volumes for Immediate Dismantlement of the Reference Mixed Oxide Fuel Fabrication Plant

Waste Type	Volume, m^3(a)	Waste Description	Radionuclide Content(b) Fraction	Waste Disposition
Compactable trash and combustible waste	25	HEPA filters	.4	Packaged, shipped for treatment
	90	Combustible trash	.01	Treated, packaged, shipped for disposal
	10	Combustible trash	.001	Packaged, shipped for treatment
Noncompactable, noncombustible waste	300	Equipment and structural materials	.2	Packaged, shipped for disposal
	1,800	Equipment and structural materials	.21	Packaged, shipped for disposal
	300	Equipment and structural materials	.02	Packaged, shipped for disposal
Concentrated liquids, wet wastes, particulate solids	200	Combustible wet wastes	.16	Treated, packaged, shipped for disposal
	100	Noncombustible wet wastes	.001	Treated, packaged, shipped for disposal

a. All volumes are raw waste before treatment or packaging.
b. Fraction of total activity shown in shutdown column of Table 8.6.1 in that waste stream.

jackhammering, drilling, and rock splitting and explosive removal of concrete; and plasma torch cutting of stainless steel equipment. Special procedures may reduce the amount of these airborne materials that reach the ventilation system. Water sprays may be used to reduce dust generation during drilling, jackhammering, rock splitting, or explosive blasting of concrete, or concrete surfaces may be precoated with contamination-fixing agents. Portable fume hoods may be positioned over equipment being cut with plasma torches. Sections of stainless steel liners may be chemically decontaminated before being cut. Glove boxes and ventilation ductwork may be filled with foam before being removed. Nevertheless, a small fraction of the airborne radioactive materials may be released to the atmosphere through the facility stack after they pass through the facility ventilation and filtration systems.

Although dismantling operations could result in more material being made airborne inside the facility than during normal operations and maintenance activities, the contamination control methods mentioned above should result in airborne releases that are equal to or less than routine releases during plant operations. It was estimated[6] that 10^{-9} of the throughput of a reference MOX FFP is released to the atmosphere from the facility ventilation system during plant operation. Applying this factor to the residual material remaining in the plant when decommissioning begins results in a total routine airborne release of 700 μg of mixed oxide (about 30 μg plutonium and 670 μg uranium) during the dismantling operation. This release would represent 10^{-9} times the radioactivity in the shutdown column of Table 8.8.6.1. No significant routine aqueous releases are expected from the MOX FFP dismantling operations.

8.6.2.5 Cost Estimate

The estimated costs for dismantling the reference MOX FFP are presented in Table 8.6.4. These costs have been derived using assumptions consistent with the cost estimates developed in a recent FRP decommissioning study.[7] The procedures outlined assume that the dismantlement proceeds with a minimum of difficulties. A 25% contingency is included in the cost estimate to account for the costs of dealing with unforeseen circumstances. Costs are presented in mid-1976 dollars. Waste transportation and disposal costs are being considered elsewhere in this study and are not included in the table.

TABLE 8.6.4. Estimated Costs for Dismantling the Reference Mixed Oxide Fuel Fabrication Plant

Cost Element	Cost, 1000s of Mid-1976 dollars
Labor	4,000
Equipment and supplies	1,000
Utilities and services	400
Demolition and site restoration	500
Owner costs	1,500
Contingency	1,500
Total	9,000

Labor costs indicated in the table include direct wages, fringe benefits, and other direct employer expenses. The allowances for utilities and services is to cover the costs of electricity, boiler fuels, water and water treatment, sewage treatment or disposal, and other utilities and services required during the decommissioning operation. Equipment costs include development and design costs for specialized decommissioning equipment, allowances for other equipment that must be purchased, as well as expendable equipment and supplies and waste packaging. Owner costs include expenses incurred directly by the owner during planning and performance of the decommissioning operations. These costs include overheads such as engineering, operating and accounting expenses, insurance, taxes, permits, licenses, and the cost of making reports and applications to regulatory agencies.

8.6.2.6 Nonradiological Impacts

The nonradiological impacts of dismantling the reference MOX FFP are discussed briefly below. Where it has not been possible to quantify the impacts, they have been related to similar impacts during construction or operation of the facility. In general, the decommissioning impacts are expected to be less than similar impacts during construction and operation.

Land Use. The facility site (4 km^2) is maintained during the time that the dismantling operation is being carried out. When the dismantling has been completed and the facility license is terminated, the site may be released for alternative uses as the owner desires. Land would be required in a Federal repository to dispose of the radioactive decommissioning wastes.

Water Use. Requirements for water during decommissioning are reduced from requirements during plant operations. Sanitary water usage is less since the decommissioning work force is less than the work force during plant operations. Requirements for process water and cooling water are virtually eliminated after the plant has been shut down, except for water used in the waste treatment systems.

Equipment and Materials. Some materials and equipment used for decommissioning become radioactively contaminated and are removed from further use following the decommissioning operations. These include steel radioactive waste disposal containers; cloth, paper, plastic, and wood used for personnel protection and contamination control; and specialized decommissioning equipment such as cutting torches, hand tools, and rock splitters. The approximate amounts of material used for this decommissioning mode are as follows: steel shipping containers, 450 MT; paper, wood, plastic, 50 MT; and equipment (mostly steel), 100 MT.

Energy. Electricity usage during decommissioning should be less than during plant operations. The primary use of electricity during dismantlement is for operating the ventilation system (about 10 MW-hr).

Diesel fuel and gasoline would be burned in plant vehicles and equipment used during decommissioning. Heavy equipment used during the demolition and site restoration phase of dismantling would be the primary consumer of vehicular fuels.

Air Quality Effects. During the demolition and site restoration phase of dismantling, quantities of fugitive dust would be generated from building demolition, loading of concrete

rubble, grading, and backfilling operations. Vehicle effluents would also be produced. The effects would be transitory and confined to the immediate site vicinity. Areas to be graded or excavated would usually be wet down to minimize dust generation.

Noise Impacts. Noise caused by decommissioning would vary with day-to-day work schedules, weather conditions, and other factors. During dismantling, the noisiest phase of work would be the demolition and site restoration phase, which may take about six months. Noise levels during these operations are expected to be comparable to the noise levels during facility construction.

Effects on Nearby Communities. The major impact on local communities would be associated with changes in the work force during the decommissioning. The work force at the plant would be reduced from about 200 people during normal operations to about 75 during the dismantling operations. The work force would be gradually reduced to zero at the end of the dismantling operations.

8.6.3 Hardened Safe Storage of the Reference Mixed Oxide Fuel Fabrication Plant

To place the reference MOX FFP in hardened safe storage all residual radioactivity in the facility is confined to portions of the plant that are separated from the remainder of the facility and the environment by permanent physical barriers. For this study, the facility is assumed to be maintained in this condition for an extended period (perhaps as long as 1000 years). Activities are required at the end of the continuing care period to terminate the facility license. These activities would include removal of all residual radioactivity above limits designated by regulatory agencies for public access to the facility. Significant refurbishing of the facility could be required to permit these activities to be carried out safely. Because most of the radioactive materials remaining in the facility at shutdown have relatively long half-lives, the final decommissioning operations will be similar in scope to the dismantlement operations described previously.

8.6.3.1 Decommissioning Plan and Schedule of Events

A description of the general types of activities required to place a retired nuclear facility in hardened safe storage was given in Section 8.2.2. An approximate schedule of events for hardened safe storage of the reference MOX FFP is presented in Figure 8.6.2.

Planning and preparation takes place during the final two years of facility operation. After facility shutdown activities are complete, equipment in the main process areas and process support areas is deactivated. Contaminated equipment and structural materials in the process support areas are removed and placed in the main process areas.

Following deactivation of the installed waste treatment equipment, the main process areas and the waste treatment areas are isolated from the remainder of the facility and the environment by the installation of permanent physical barriers. Any radioactive contamination remaining outside the isolation structure is removed, packaged, and shipped offsite.

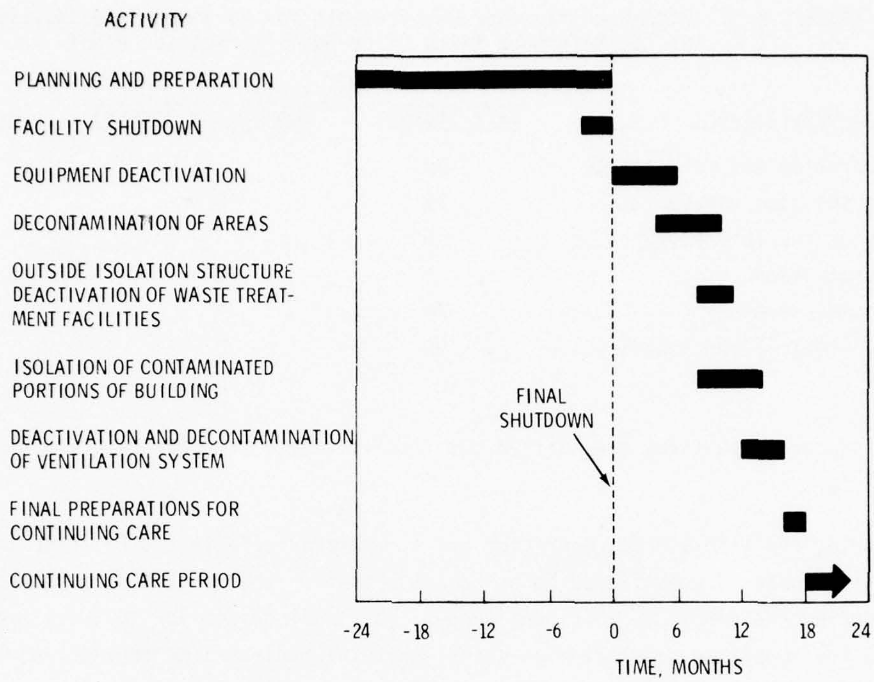


FIGURE 8.6.2. Approximate Schedule of Events for Hardened Safe Storage of the Reference Mixed Oxide Fuel Fabrication Plant

The ventilation system is the final portion of the facility to be deactivated. All contaminated equipment, filters, and ductwork are packaged and shipped offsite for disposal. Most exterior doors into the facility are disabled. The exclusion area fence is maintained, and the remainder of the site is assumed to be released for alternate uses. Activities during the continuing care period include environmental monitoring and inspection of the isolation structures. The facility is protected from unauthorized entry by the exclusion fence and locked or disabled exterior doors.

8.6.3.2 Manpower Requirements

The estimated manpower requirements for hardened safe storage of the MOX FFP are presented in Table 8.6.5. The manpower requirements for one year of continuing care and final decommissioning are also given in the table. Continuing care activities are assumed to be carried out indefinitely.

8.6.3.3 Radioactive Wastes

Most of the radioactivity in the MOX FFP at shutdown remains in the facility. However, there are some wastes generated during the safe storage operations that would require offsite disposal. These wastes are composed primarily of waste generated in the final cleanup operations after the isolation structures are closed, and of combustible wastes generated after the onsite combustible waste treatment facility is decommissioned. Combustible and wet wastes generated during the first phases of the decommissioning are assumed to be converted to a nonflammable solid in the installed waste treatment facilities and placed in the isolation structure in nonflammable containers. The estimated volumes of waste requiring offsite

TABLE 8.6.5. Estimated Manpower Requirements for Hardened Safe Storage of the Reference Mixed Oxide Fuel Fabrication Plant

Job Description	Man-Years Safe Storage	Man-Years Continuing Care(a)	Final Decommissioning
Facility operation and maintenance	20		20
Engineering and supervisory	15	0.1	25
Health physics and laboratory	5		12
Radiation protection	5	0.1	8
Skilled decommissioning	25		45
Management, clerical and security	30	0.1	50
Total	100	0.3	100

a. Continuing care activities are carried out indefinitely. Manpower requirements per year are shown.

disposal during the safe storage operation are summarized in Table 8.6.6. Less than one m^3/yr of radioactive wastes are estimated to be generated during the continuing care period. Waste generated during the final decommissioning operations will be similar to those presented in Table 8.6.3 for immediate dismantlement except that the radionuclide content should be based on the 1000 year column in Table 8.6.1.

TABLE 8.6.6. Estimated Waste Volumes from Hardened Safe Storage of the Reference Mixed Oxide Fuel Fabrication Plant

Waste Type	Volume, m^3	Waste Description	Radionuclide Content Fraction(a)	Waste Disposition
Compactable trash and combustible waste	25	HEPA filters	.4	Packaged, shipped for treatment
	10	Combustible trash	6×10^{-4}	Packaged, shipped for treatment
Noncompactable, noncombustible waste	60	Equipment and structural materials	.02	Packaged, shipped for treatment

a. Fraction of activity shown in shutdown column of Table 8.6.1 in that waste stream.

Combustible wastes generated during the latter stages of the entombment operation are typical of combustible wastes generated during plant operations and maintenance activities. They are composed of protective clothing, plastic and cloth coverings, and wood.

Noncombustible wastes shipped offsite during the entombment operations are primarily produced from the decontamination of the ventilation system. They include HEPA filters, stainless steel ductwork and equipment, and concrete rubble. Estimated radioactive contamination levels on wastes generated during the entombment operations are also included in the table. Radioactive contamination levels on the waste are expected to be typical of operating wastes from the facility.

8.6.3.4 Routine Radioactive Effluents

Routine radioactive effluents during hardened safe storage of the MOX FFP are primarily airborne releases of gases and particles. No significant routine aqueous releases are expected during the entombment operation. Airborne releases are expected to be less than releases during dismantlement, since fewer decommissioning activities are carried out that could cause radioactive material to become airborne in the facility. Contamination control methods utilized during operations could result in airborne contamination inside the facility. Based on the release predicted for dismantlement of the MOX FFP, it is estimated that the routine airborne radioactive effluent from entombing the reference MOX FFP would have a radioactivity content equivalent to 10^{-11} times the shutdown column of Table 8.6.1.

8.6.3.5 Cost Estimate

The estimated costs for hardened safe storage of the reference MOX FFP are presented in Table 8.6.7. The assumptions, definitions and limitations associated with this cost estimate are similar to those presented for the dismantlement mode. The estimated annual cost of continuing care and the final decommissioning cost are also indicated in the table. Most of the cost of continuing care is associated with owner costs such as taxes and insurance. Continuing care activities at the facility are assumed to be carried out for periods up to 1000 years.

TABLE 8.6.7. Estimated Costs for Hardened Safe Storage of the Reference Mixed Oxide Fuel Fabrication Plant

	Cost, 1000s of Mid-1976 Dollars		
Cost Element	Safe Storage	Continuing Care Per Yr	Final Decommissioning
Labor	2,500	8	3,800
Equipment and supplies	500	5	700
Utilities and services	300	2	200
Owner costs	700	10	1,200
Contingency	700	5	1,200
Total	5,000	30	7,000

8.6.3.6 Nonradiological Impacts

The nonradiological impacts of hardened safe storage of the reference MOX FFP are discussed briefly below.* Where it has not been possible to quantify the impact, they have been related to similar impacts during construction or operation of the facility. In general, the decommissioning impacts are expected to be less than similar impacts during construction or normal operation of the facility.

* Impacts during the final decommissioning operations at the end of the continuing care period have not been included. These impacts are similar to those described previously for immediate dismantlement.

Land Use. The 0.06 km^2 (15 acre) exclusion area of the reference MOX FFP site is occupied indefinitely by the facility. The remaining 4 km^2 of the site can be released for alternate use at the discretion of the facility owner. Activities near the exclusion area are restricted to those that would not jeopardize the integrity of the entombed structure.

Water Use. Requirements for water during decommissioning are reduced from requirements during plant operations. Sanitary water usage is less since the decommissioning work force is less than the work force during plant operations. Requirements for process water and cooling water are virtually eliminated after the plant has been shut down (except for water used in the waste treatment systems).

Equipment and Materials. Some materials and equipment used for decommissioning become radioactively contaminated and are removed from further use following the decommissioning operations. This includes steel radioactive waste disposal containers; cloth, paper, plastic, and wood used for personnel protection and contamination control; and specialized decommissioning equipment such as cutting torches, temporary shielding, hand tools and rock splitters. The approximate amounts of material used for this decommissioning mode is as follows: steel shipping containers, 50 MT; paper, wood, plastic, 20 MT; and equipment (mostly steel), 5 MT.

Energy. Electricity usage during decommissioning should be less than during plant operations. The primary use of electricity during decommissioning is for operating the ventilation system (about 5 MW-hr for entombment). Diesel fuel and gasoline would be burned in plant vehicles and equipment used during decommissioning. Fossil fuels would be consumed for space heating. The amount of vehicular and other fossil fuels consumed during decommissioning is expected to be less than the amounts consumed during plant operations.

Air Quality Effects. The safe storage operations are not expected to noticeably affect air quality near the site. The primary source of effluents will be from vehicular traffic. Vehicular traffic during the safe storage operations is expected to be less than the traffic during plant operations.

Noise Impacts. Noise caused by decommissioning would vary with day-to-day work schedules, weather conditions and other factors. The primary source of noise during the safe storage operations would be vehicular traffic. As mentioned previously, vehicular traffic is expected to be less than during plant operations.

Effects on Nearby Communities. The major impact on local communities would be associated with changes in the work force at the plant. The work force would be reduced from about 200 people during normal operations to about 50 people during the decommissioning operations to the equivalent of less than one person during the perpetual continuing care period.

Institutional Effects. The continuing care activities at the entombed facility would continue for at least several hundred years and perhaps longer. Institutions do not exist to assure that the structure remains in a condition that would pose minimum risks to the public and the environment for this length of time.

8.6.4 Selection of a Decommissioning Alternative for the Mixed-Oxide Fuel Fabrication Plant

Immediate dismantlement was selected as the reference alternative for decommissioning the MOX FFP. Significant quantities of long-lived radioactive contamination are expected to remain in the facility at the end of its operating lifetime. Many uncertainties exist for hardened safe storage of a facility containing long-lived radioactive elements. Dismantlement removes the radioactivity to a deep geologic repository, releasing the MOX FFP site for other uses.

8.7 ACCIDENTS DURING DECOMMISSIONING

8.7 ACCIDENTS DURING DECOMMISSIONING

Scenarios have not been developed for accidents that might occur during decommissioning activities at the four reference facilities. It is felt that the frequency and overall consequences of accidents resulting in public radiation exposure would be much less for decommissioning activities than for accidents during normal plant operations. Considerations that lead to this conclusion are outlined below.

- The safety aspects of decommissioning a fuel reprocessing plant were investigated in detail in Reference 7. Public radiation exposure from accidents during decommissioning was found to be quite low. A similar conclusion for decommissioning activities at all fuel cycle facilities was reached in Reference 3.
- Inventories of radioactive materials in a facility when decommissioning begins are typically many orders of magnitude less than during plant operations. For example, all spent fuel, plutonium and uranium product material, and process wastes are removed from a reprocessing plant before decommissioning activities begin. All spent fuel is removed from reactors and independent spent fuel storage facilities and all plutonium and uranium feed materials and fuel bundles are removed from a mixed oxide fuel fabrication facility.
- Radioactive material remaining in the facility when decommissioning begins is generally in a relatively nondispersable form. Much of the radioactivity is bound to structural materials or to the inside surfaces of equipment and piping. In many instances, the radioactive material remaining in a decommissioned facility has resisted removal by strong chemical solutions used in the initial stages of the decommissioning operations.
- All safety systems (such as filtered ventilation systems) remain in operation while the decommissioning activities are being carried out. Even if an accidental release occurred inside the facility during decommissioning, a very small fraction of the radioactive material would be released after passing through these systems.
- The accident initiating mechanisms that remain in a facility after shutdown are generally capable of producing much lower accident forces than the mechanisms presented during plant operations.
- Many of the plant systems with potential for producing an accident with offsite consequences have been shut down and no longer represent a safety concern during decommissioning.

It should be noted that some activities carried out at a facility during the decommissioning are typical of similar activities carried out during plant operations and have similar potential for producing accidents. The most important of these activities is operation of the onsite waste treatment facilities to treat waste generated during the decommissioning activities. Accident descriptions for these facilities are presented elsewhere in this document. Because of the short duration of the decommissioning activities relative to the plant operation lifetime, accidents in the waste treatment facilities during decommissioning are not expected to be an important safety issue.

8.8 PHYSICAL PROTECTION AND SAFEGUARDS REQUIREMENTS FOR DECOMMISSIONING

8.8 PHYSICAL PROTECTION AND SAFEGUARDS REQUIREMENTS FOR DECOMMISSIONING

Prior to decommissioning a nuclear facility the licensee would prepare and obtain NRC approval of a decommissioning plan. The plan would contain descriptions of the physical protection and safeguards procedures that would be followed during decommissioning to insure that the public health and safety would not be threatened by acts of sabotage or theft and illicit use of radioactive and special nuclear material. Although all nuclear and radioactive materials would be removed from the facility and a cleanout of the processing equipment and processing areas would be carried out prior to decommissioning, some radioactive material may normally remain essentially fixed as contaminants on equipment and structural material. Most of the material removed during chemical decontamination and physical dismantling will be collected in leach solutions. In addition, there is the possiblity that substantial quantities of special nuclear material (SNM) or other radioactive material would be recovered from previously "inaccessible" places when the structural and equipment items are dismantled. The physical security and safeguards plan would take account of this possiblity and provide for appropriate physical protection, control and accountability of such material in the event the quantities isolated at one time and place are sufficient to justify concern for safeguards.

The present NRC regulations provide guidelines for the degree of physical protection and other safeguard measures that would be needed. All licensed nuclear facilities must be protected against industrial sabotage and theft of SNM in accordance with Section 73.40 of 10 CFR 73. As long as radioactively hazardous materials are present, physical protection of the facility would be required. A locked, intrusion-resistant enclosure for the facility would probably be sufficient when the licensee can provide assurance that the quantities of SNM present are insignificant.

If the quantity of SNM present at one time and place were to exceed one effective kilogram as defined in Section 70.22 of 10 CFR 70, a variety of additional material control and accountability requirements would be necessary, as described in 10 CFR 70. If the quantity of strategic special nuclear material (e.g., plutonium) present at one time and place were to exceed five formula kilograms, as defined in Section 73.50 of 10 CFR 73, the additional physical protection requirements specified in Section 73.50 and 73.60 would probably be necessary (see 3.9.3 for a summary of these requirements). However, if all of the SNM is widely distributed throughout the plant in equipment and structural materials, as would usually be in the case during a decommissioning operation, the licensee may be able to obtain an exemption from some of the physical protection and safeguard requirements such as those specified for operating licensees who possess and use greater than the five formula kilograms and one effective kilogram threshold quantities.

Because decommissioning would result in decontamination and isolation of all radioactive materials and controlled disposal of these materials as wastes, no significant safeguard problems would arise that could not be adequately controlled by the safeguard requirements which would be specified in the license agreement for decommissioning. The current physical protection and safeguard procedures can be adapted to decommissioning and are believed to be sufficiently definitive to satisfy the requirements.

REFERENCES FOR SECTION 8.0

1. Title 10, Code of Federal Regulations, Part 20 (10 CFR 20).

2. Alternatives for Managing Wastes from Reactors and Post-Fission Operations to the LWR Fuel Cycle, ERDA 76-43, U.S. Energy Research and Development Administration, Washington DC, May 1976, Chapter 15.

3. Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle, NUREG-0116, U.S. Nuclear Regulatory Agency, Washington DC, October 1976, pp 4-129 thru 4-143.

4. J. F. Nemec and K. G. Anderson, Demolition of Radioactive and Contaminated Concrete Structures by Use of Explosives, Presented at the 1974 Annual Meeting of the American Nuclear Society, Philadelphia, PA, June 1974.

5. W. J. Bair, Pacific Northwest Laboratory Annual Report for 1976, (BNWL-2100 PT5), Battelle, Pacific Northwest Laboratories, Richland, WA, June 1977, p 26.

6. W. J. Manion and T. S. Laguardia, An Engineering Evaluation of Nuclear Power Reactor Decommissioning, AIF/NESP-009, National Environmental Studies Project, Atomic Industrial Forum, Washington DC, November 1976.

7. K. J. Schneider, C. E. Jenkins et al., Technology Safety and Costs of Decommissioning a Reference Fuel Reprocessing Plant, NUREG-0278, prepared for the U,S. Nuclear Regulatory Commission by Battelle, Pacific Northwest Laboratories, Richland WA, September 1977.

8. Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Reactors, NUREG-0002, U.S. Nuclear Regulatory Commission, Washington DC, August 1976, pp. 11, 27.

9.0 WASTES FROM THORIUM FUEL CYCLES

9.0 WASTES FROM THORIUM FUEL CYCLES

Thorium is a fertile material which can play essentially the same role that uranium-238 plays in commercial LWRs. There are four times as much thorium in the earth's crust as uranium; however, commercialization of the thorium cycle has not to date been economically attractive, in large part because in nature, thorium does not contain a fissile specie like uranium, which contains uranium-235. Even so, thorium and uranium contained in a ton of earth of average composition would be equivalent to the heat in 20 tons of coal.

There is a striking parallel between the thorium and the uranium series (Figure 9.0.1)[1]: both series begin with an abundant fertile material, thorium-232 or uranium-238, whereupon neutron capture forms a new isotope that beta decays to form a fissile isotope, uranium-233 or plutonium-239; thereafter, each series has the same rhythm of fertile isotope and parasite, which terminates each chain with a chemically separable element. For thorium fuel cycles, the major fissile product is uranium-233, some of which fissions in place and the remainder of which can be separated from the fertile material by chemical reprocessing and can then be inserted back into the core after fabrication into fuel elements. Plutonium plays the same role when recycled in uranium-238.

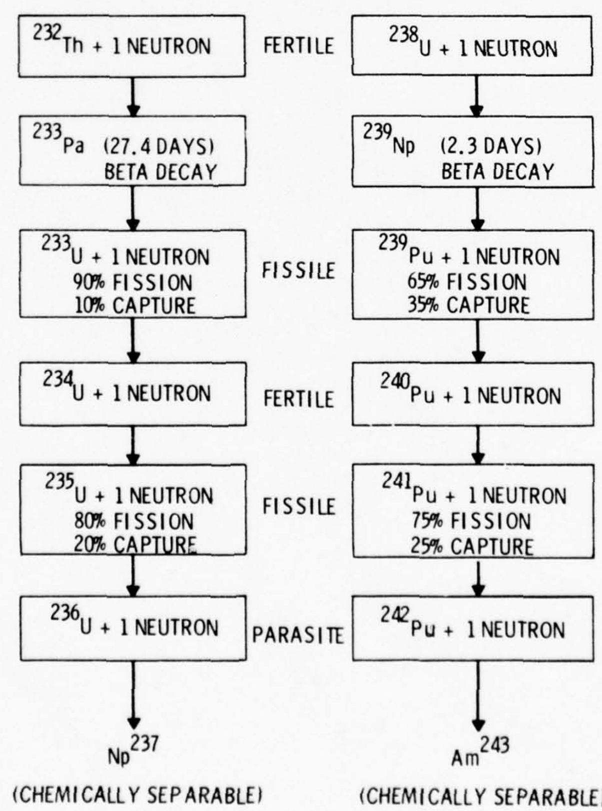


FIGURE 9.0.1. Comparison of Thorium and Uranium Series

If thorium were commercialized in the near future, the earliest uses in the United States would likely be in a high temperature gas-cooled reactor (HTGR)[2] and in a light water reactor (LWR) pioneered by the light water breeder reactor (LWBR)[3] system. Prototypes of each system have been licensed in the U.S.A. Already the Fort St. Vrain Public Service of Colorado has a 300 MWe HTGR progressing to full power. An LWBR-type core has been installed in the Shippingport Reactor of Duquesne Light and Power Company. The Peach Bottom plant of Philadelphia Electric (now mothballed) was an early prototype of General Atomic HTGRs.

In this section, a qualitative comparison is made of wastes generated by the uranium fuel cycle and the thorium fuel cycle. This study presents data characterizing wastes from prebreeder light water breeder reactors using thorium and slightly enriched uranium-235. The prebreeder LWBRs are essentially LWRs using thorium. The operation of HTGR and LWBR cycles are conceptually designed, and wastes produced in these cycles are compared for potential differences.

It appears that long-term waste characteristics of the thorium cycle are essentially the same as those for the uranium-238 cycle. Left to economic factors alone,[4] thorium probably would not be used for many years. However, if other constraints are applied, thorium may be utilized within 20 years or so.

9.1 THORIUM AS A REACTOR FUEL

9.1 THORIUM AS A REACTOR FUEL

Thorium can be used as a fertile material in almost any type of reactor although the bred product, uranium-233, is advantageous as fuel for thermal reactors rather than for fast reactors. Thorium and uranium-233 have been suggested for use in: 1) LWRs without attempting breeding,[5] as well as in LWBRs, 2) nonbreeding high temperature gas-cooled reactors (HTGRs),[6,7] 3) near breeding CANDU reactors,[8,9] and 4) fast breeders, in particular, fast converter reactors.[10] Studies are being conducted on the Molten Salt Reactor (MSRE). Only the HTGR thorium cycle and the LWBR are discussed here since other fuel cycles are not as fully developed. However, the waste characteristics of thorium fueled LWRs are essentially the same as those of LWBRs.

9.1.1 Operational Modes

In principle, thorium-fueled reactors can be operated in either a recycle or nonrecycle mode. In the recycle mode, spent fuel is reprocessed to separate the fissile material (uranium-233) which has been generated or which remains unburned from the previous irradiation. This fissile material can then be refabricated into fuel elements for reinsertion into the core. This can be done whether or not the amount of fissile material generated is large enough so that the reactor constitutes a true breeder, which once started provides all its own fissile material.

In the nonrecycle mode, the fissile material generated is not returned to a core, either because the fuel is not reprocessed or because the product from the reprocessing plant is treated as waste or is stored for future use. Since uranium-233 does not occur naturally, all reactors which use it must effectively be started up using fissile material isolated by chemical or isotopic processing. Thus, it is unlikely that the nonrecycle mode would be utilized significantly for thorium fuels.

In the environmental statement for the LWBR,[11] the startup reactor is referred to as a "prebreeder." The prebreeder includes fertile thorium to generate fissile uranium-233 and an initial fissile charge, which could be either slightly enriched or fully enriched uranium-235, or plutonium.

Similar considerations apply to the HTGR. Even though equilibrium cycle HTGRs (which do not qualify as breeders) use recycle uranium-233, they must be supplemented as well as started up using some other fissile material or uranium-233 from some other source. In this report, HTGR data are presented only for equilibrium cycles after startup, using recycled uranium-233 plus uranium-235 supplement.

The fuel cycle details influence the characteristics of the reprocessing waste, since they determine what is waste and what is recycled. In the case of the LWBR and pre-LWBR, enough plutonium is generated to be usefully recovered for its energy value; thus, it is assumed here that plutonium is recovered during reprocessing and only 0.5% appears in the waste. In the case of the equilibrium HTGR, the amount of plutonium generated is quite small from a fuel standpoint, so it is generally assumed (and will be here) that plutonium is not recovered in reprocessing, but is included in the high-level waste. Consequently,

there is more plutonium in the reprocessing waste from HTGRs than from pre-LWBRs or LWRs, even though the latter two generate more plutonium during irradiation in the core. Reclaiming plutonium from spent HTGR fuel could change this situation. The denatured thorium cycle[12] proposed for LWRs also produces relatively large amounts of plutonium from a waste storage viewpoint. Possibly such plutonium would also be recycled.

9.2 HIGH TEMPERATURE GAS-COOLED REACTOR FUEL CYCLE DESCRIPTIONS

9.2 HIGH TEMPERATURE GAS-COOLED REACTOR FUEL CYCLE DESCRIPTIONS

The high temperature gas-cooled reactor (HTGR) uses helium as the coolant and is classified as an advanced converter when operating on the thorium/uranium cycle. Uranium enriched in uranium-235 is the fissile material in the initial core and in makeup fuel elements. Uranium-233 bred in the thorium is fissile material which is to be recovered and reused, substantially reducing the number of makeup fuel elements. Fissile uranium may be either highly enriched (HEU) or low enriched (LEU). In both fuel cycles, approximately one-fourth of the fuel is discharged each year.

9.2.1 Highly Enriched Uranium and Low Enriched Uranium Fuel Cycles

The HEU-HTGR fuel cycle uses thorium and 93% enriched uranium-235 in the initial fuel and in the makeup fuel. Uranium-233 (70 to 80% isotopically pure) bred in the thorium is recovered from the spent fuel, fabricated into recycle fuel elements and returned to the reactor. Sufficient unburned uranium-235 remains in spent initial and makeup fuel to warrant recovery and recycle to the reactor an additional cycle prior to retirement and disposal as waste. The particle configuration of the fuel permits physical separation of the initial uranium from the thorium containing bred uranium-233. Unburned uranium-233 in the recycle element is recovered for reuse along with bred uranium-233. From the standpoint of radioactive waste, the plutonium generated is not trivial. Moreover, plutonium-238 found via uranium-235 decays to uranium-234, a long-term source of radium-226. Figure 9.2.1 illustrates this fuel cycle. Table 9.2.1 defines the average fuel elements.

The LEU-HTGR fuel cycle uses thorium and uranium enriched to 20% uranium-235 in the initial core and in the makeup fuel. Uranium-233 in the thorium is recovered from the spent fuel, refabricated into recycle fuel elements and returned to the reactor. Prior to decontamination in the reprocessing plant, the bred uranium is denatured to 15% uranium-233 by the addition of uranium-238 to reduce the possibility of unauthorized diversion. The unburned uranium-235 and uranium-233, which include uranium-234 and uranium-236 along with substantial amounts of plutonium, are not recovered. Plutonium is bred in uranium-238 in the TRISCO-coated fissile particles. The particles are packaged for long-term storage or are discarded as waste. Figure 9.2.2 illustrates the LEU-HTGR fuel cycle. Table 9.2.2 defines the average fuel elements.

The LEU and HEU fuels used in this analysis differ from each other in other respects. The LEU fuel is designed for a lower power density; consequently, it has approximately 15% more fuel elements per GWe-yr. Because of the lower enrichment, the spent fuel contains more uranium and transuranic actinides. However, compared with the LEU wastes, the HEU wastes may contain more uranium-234 from plutonium-238 because of larger amounts of neptunium-237. This results from uranium-236 formed from uranium-235 exacerabated by recycle of uranium-235 (and thus uranium-236) from previous fuel loads. The uranium-234 content via capture in uranium-233 and protactinium-233 is nearly the same for most thorium cycles. Moreover, protactinium-231 levels are the same for the LEU and HEU cycles; but these levels are greater for the LWBR because of the n-2n reaction on thorium. Protactinium-231 levels could be reduced by making uranium-232 from it or by burning it out.

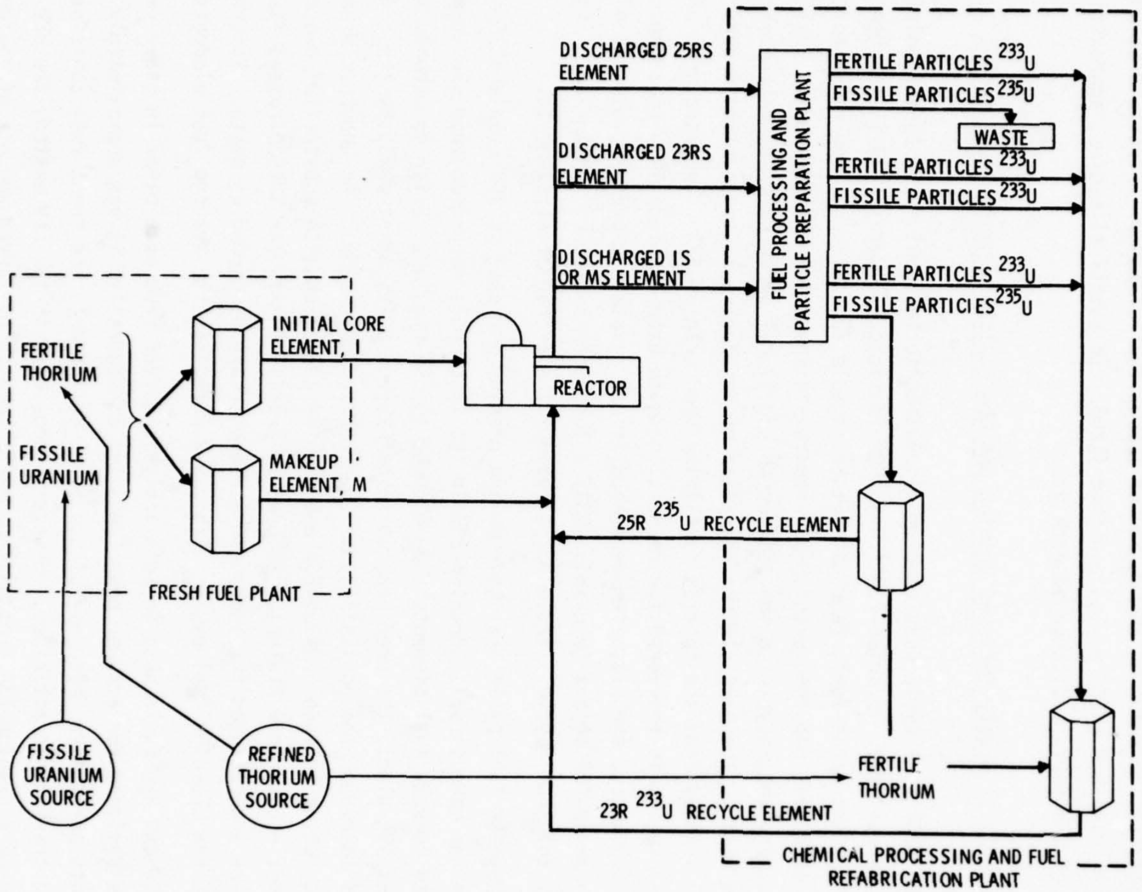


FIGURE 9.2.1. Highly Enriched Uranium - High Temperature Gas-Cooled Reactor Fuel Cycle

TABLE 9.2.1. Average Highly Enriched Uranium-High Temperature Gas-Cooled Reactor Spent Fuel Element

Component	Type of Spent Fuel Element (kg per fuel element) IM	23R	25R
Graphite block, including dowels and fuel hole plugs	84.6	84.6	84.6
Fissile particles			
Kernel	Resin UOC_4	Resin UOC_4	Resin UOC_4
Coating	TRISCO	TRISCO	TRISCO
Total U	0.220	0.160	1.40
Other heavy metals	0.025	0.003	0.17
Fission products	0.581	0.502	0.43
Carbon and oxygen	0.179	0.147	0.45
Total kernel	1.005	0.812	2.45
SiC coating	2.10	1.32	4.03
Carbon coatings	3.32	4.25	6.49
Total particle	6.43	4.25	12.97
Fertile particles			
Kernel	ThO_2	ThO_2	ThO_2
Coating	BISO	BISO	BISO
Total U	0.264	0.264	0.264
Total Th	7.936	7.936	7.936
Other heavy metals	0.173	0.173	0.173
Fission products	0.334	0.334	0.334
Oxygen	1.036	1.036	1.036
Total kernel	9.743	9.43	9.743
Carbon coatings	5.547	5.547	5.547
Total particle	15.290	15.290	15.290
Shim particle	7.92	7.92	7.92
Fuel rod matrix	3.95	3.95	3.95
Total fuel rods	33.59	31.41	40.13
Poison rods			
B_4C	0.019	0.019	0.019
Matrix	0.546	0.546	0.546
Total poison rods	0.565	0.565	0.565
Total fuel element	118.8	116.6	125.3
Number in equilibrium cycle	575	388	101

IM = Initial and makeup fuel elements containing 93% enriched U-235 in U-238
23R = Recycle fuel elements containing U-233
25R = Recycle fuel elements containing U-235
MTHM/GWe-yr = 10.2

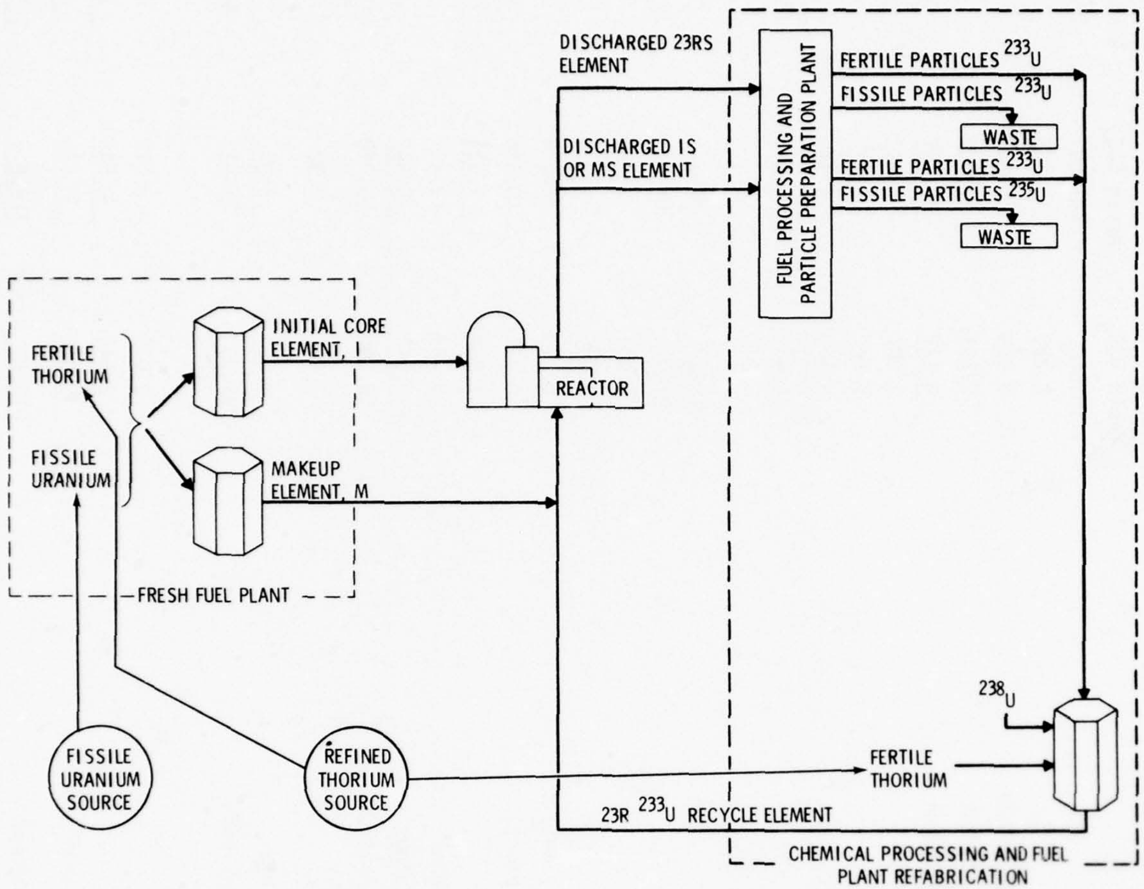


FIGURE 9.2.2. Low Enriched Uranium High Temperature Gas-Cooled Reactor Fuel Cycle

TABLE 9.2.2. Average Low Enriched Uranium-High Temperature Gas-Cooled Spent Fuel Element

Component	Type of Spent Fuel Element (kg per fuel element) IM	23R
Graphite block, including dowels and fuel hole plugs	84.6	84.6
Fissile particles		
Kernel	Resin UOC_4	Resin UOC_4
Coating	TRISCO	TRISCO
Total U	2.547	3.891
Other heavy metals	0.085	0.115
Fission products	0.581	0.535
Carbon and oxygen	0.864	1.221
Total kernel	4.077	5.762
SiC coating	6.61	9.47
Carbon coatings	10.80	15.26
Total particle	21.49	30.49
Fertile particles		
Kernel	ThO_2	ThO_2
Coating	BISO	BISO
Total U	0.158	0.158
Total Th	5.043	5.043
Other heavy metals	Present but not determined	Present but not determined
Fission products	0.143	0.143
Oxygen	0.717	0.717
Total kernel	5.918	5.918
Carbon coatings	3.369	3.369
Total particle	9.287	9.287
Shim particle	5.00	3.00
Fuel rod matrix	3.95	3.50
Total fuel rods	39.73	46.28
Poison rods		
B_4C	0.02	0.02
Matrix	0.55	0.55
Total poison rods	0.57	0.57
Total fuel element	124.9	131.5
Number in equilibrium cycle	1058	264

IM = Initial and makeup fuel elements containing 20% enriched U-235
23R = Recycle fuel elements containing U-233/U-238 (15% U-233)
MTHM/GWe-yr = 10.6

9.2.2 Fuel Element Characterization

The fuel element for HTGR is a hexagonal block of graphite approximately 790 mm (31 in.) long and 360 mm (14 in.) across the flats (Figure 9.2.3). These fuel elements are stacked in a close-packed array in the HTGR. The blocks contain drilled holes for helium coolant flow and plugged holes for the fuel.

The fuel used in the HTGR is in the form of ceramic microspheres coated with pyrolytic carbon and silicon carbide (Figure 9.2.4) bonded into fuel rods using a carbonaceous matrix. These fuel rods are placed in the fuel holes in the graphite blocks. Three types of fuel elements can be used, classified by the type of fissile materials that they contain:

1) the initial and makeup (IM) fuel element, which contains uranium-235 and thorium, and is used in initial and makeup fuel loading;

2) a uranium-233 recycle (23R) element, which is bred uranium-233 and thorium, and is the major fuel element to be refabricated; and

3) the uranium-235 recycle (25R) elements which contain uranium-235 and thorium, and are used when it is desirable to recycle uranium-235 in the HEU-HTGR.

With fuel recycle, an HTGR contains more than one type of fuel element. In addition to the fuel manufactured in a fresh fuel plant, the reactor also contains recycle fuel from the refabrication plant. For the HEU-HTGR, there are two types of recycled uranium-235. For the LEU, there is only the recycled fuel containing bred uranium-233. All fuel types contain two distinct kinds of fuel particles. The fertile particles contain thorium and the fissile particles, uranium.

9.2.3 Fuel Recycle Operations

The same basic fuel recycle operations are used for both the HEU- and LEU-HTGR fuel cycles. After a suitable decay time, the spent fuel is broken into small particles (<5 mm) in a three-stage crusher. The graphite is oxidized to CO_2 in a fluidized bed burner. The burner ash is then separated by pneumatic classification into its two major components, the SiC-coated fissile particles and the ThO_2 fertile particles. A nitric acid solution containing fluoride catalyst is used to dissolve the ThO_2. With LEU-HTGR fuel depleted, uranyl nitrate is also added to the dissolver solution for denaturing processes. An Acid-Thorex solvent extraction process (see Section 9.3.2.3) is used to separate the uranium, thorium and fission products from each other. Purified uranium is routed to the refabrication facility, recovered thorium is stored to allow the excess thorium-228 to decay, and fission products are solidified.

All of the LEU- HTGR cycle fissile particles, as well as those from recycled uranium-235 fuel elements in the HEU-HTGR fuel cycle, are packaged for disposal or long-term storage. The SiC coatings on the separated fissile particles from HEU cycle initial elements, makeup elements and 23R fuel elements are broken in a roll crusher and burned in a fluidized bed burner to oxidize the inner carbon coating and to convert the uranium oxycarbide to U_3O_8. The fissile

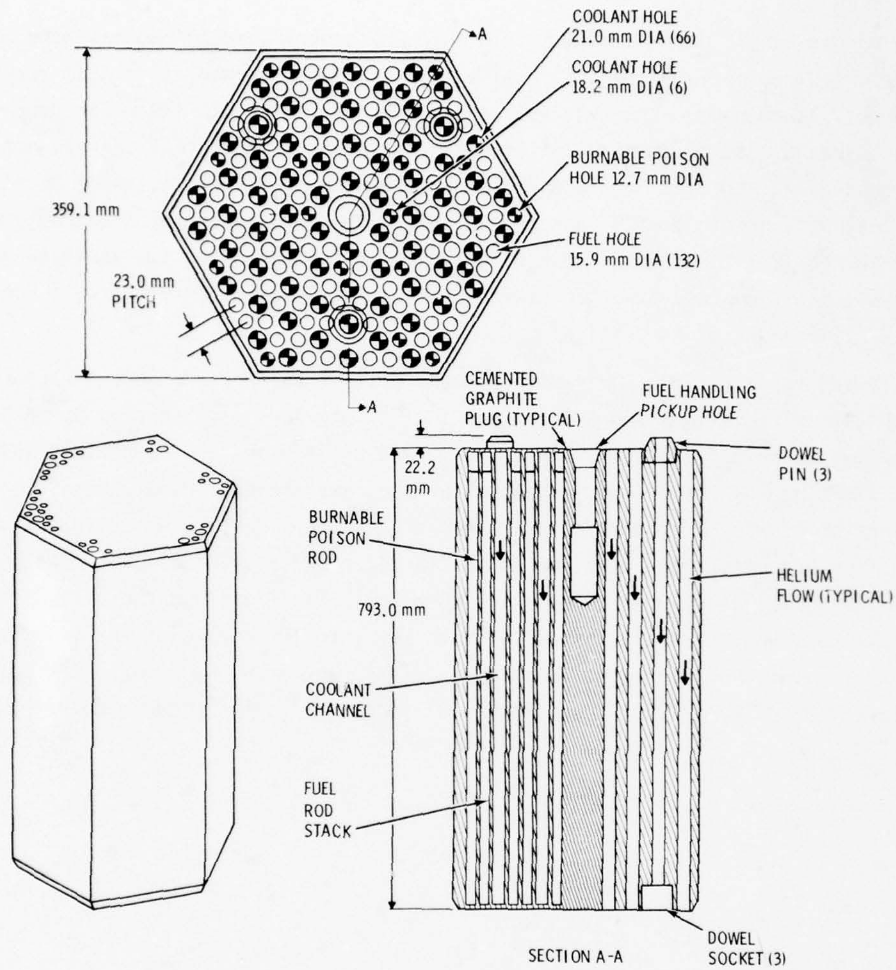


FIGURE 9.2.3. High Temperature Gas-Cooled Reactor Fuel Element

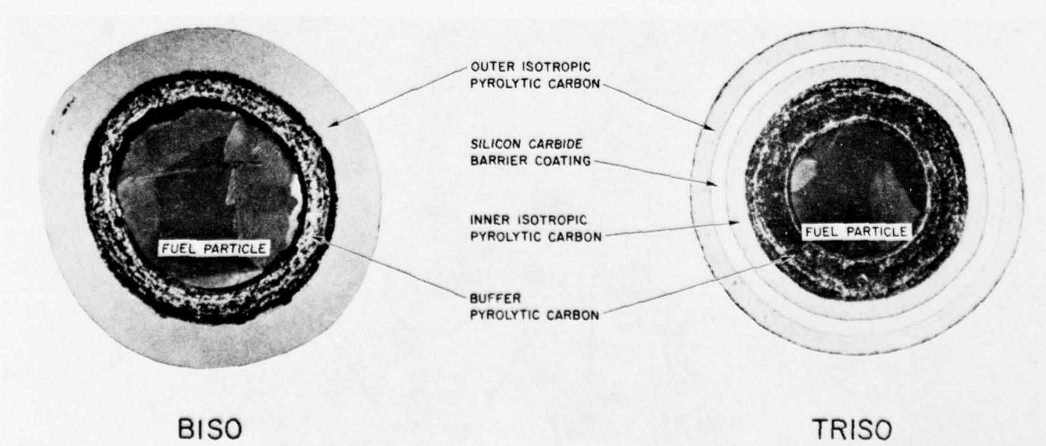


FIGURE 9.2.4. Coated Fuel Particles

particle burner ash is dissolved in nitric acid and centrifuged to separate the SiC hulls, which are dried and packaged for disposal as waste. Dissolver solution from recycled uranium-233 fuel elements is combined with that from the thorium particle dissolver for processing through the Acid-Thorex process. Uranium-235 dissolver solution from the initial and makeup fuel is decontaminated and purified by the Purex-type solvent extraction process. The fission product waste solution is combined with that from the Acid-Thorex process for treatment; then the purified uranium-235 solution is routed to the refabrication plant for separate processing into recycle fuel. Reprocessing flow sketches for the HEU- and LEU-HTGR fuels are shown in Figures 9.2.5 and 9.2.6, respectively.

Following reprocessing, the recovered uranium is transferred from the reprocessing facility to refabrication operations where it is fabricated into fuel microspheres by an ion-exchange-resin loading technique. After drying, the loaded resin beads are carbonized in fluidized bed units, converting the uranium to kernels of uranium oxycarbide. Off- gases from the carbonization unit are scrubbed with a recirculating stream of perclene ($CaCl_4$), which removes the resin decomposition products. The perclene is purified for reuse by distillation with the bottoms routed to an incinerator. The kernels are then coated in fluidized bed coaters to form layers. The various coating layers are formed by injecting into the bottom of the fluidized bed appropriate mixtures of ethylene, propylene, silane, argon and hydrogen. The coater operating temperature is 1000°C to 1200°C, which thermally decomposes the organic materials forming the coatings.

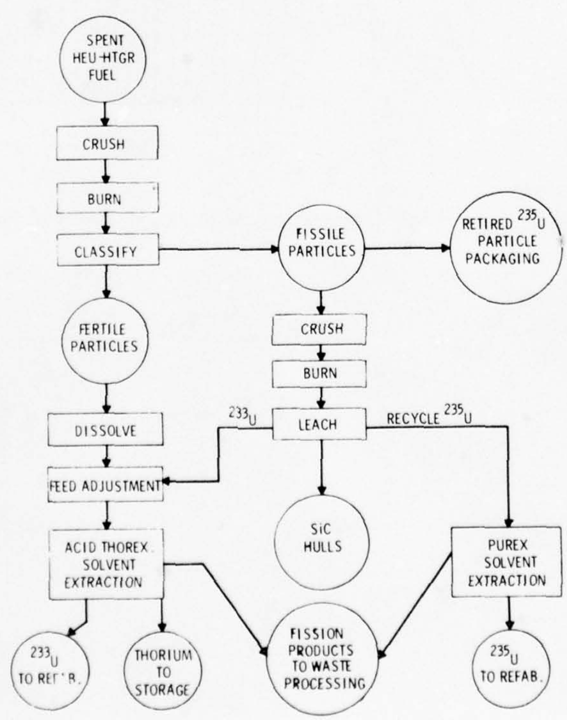


FIGURE 9.2.5. Highly Enriched Uranium - High Temperature Gas-Cooled Reactor Fuel Reprocessing

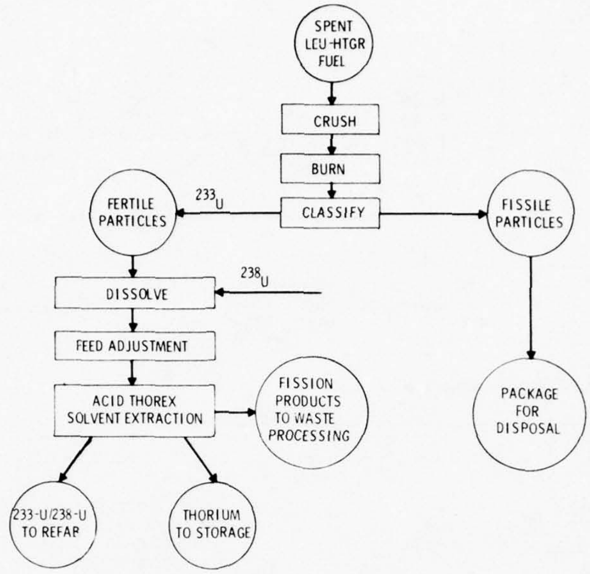


FIGURE 9.2.6 Low Enriched Uranium - High Temperature Gas-Cooled Reactor Fuel Reprocessing

The coater off gases are passed sequentially through a perclene scrubber and a caustic scrubber prior to filtration and discharge. The perclene removes soot and other carbonaceous materials and the caustic scrubber removes the HCl formed from the decomposition of the silane. The loaded perclene is recovered in a spray dryer with the dryer bottoms sent to waste processing. Periodically, the graphite liners and gas distributor frits in the coaters require replacement. They are processed through a scrap recovery system where the parts are burned, the ash leached and the recovered uranium purified in the reprocessing plant solvent extraction system.

Next, the coated particles are blended with coated thoria particles, made in a separate facility, and small graphite particles. The mixture is placed in a mold and injected with a pitch material forming the fuel rods. The fuel rods are loaded into fuel elements, carbonized, packaged, and shipped to a reactor. The off-gases from the fuel rod carbonization step are treated in the same manner as those from the resin carbonization step. Refabrication reject material and scrap are specially treated for maximum internal recycle within the reprocessing and refabrication processes. In addition to these primary fabrication operations, numerous inspections are conducted to ensure the quality of the product, to enable process control, and to provide for material accountability.

The uranium-233 from both the HEU and LEU fuel cycles, as well as the recycle uranium-235 from the HEU cycle, is refabricated by the same process. Separate processing lines are used for the two different fissile uranium isotopes. Figure 9.2.7 is a flow sketch of the refabrication process.

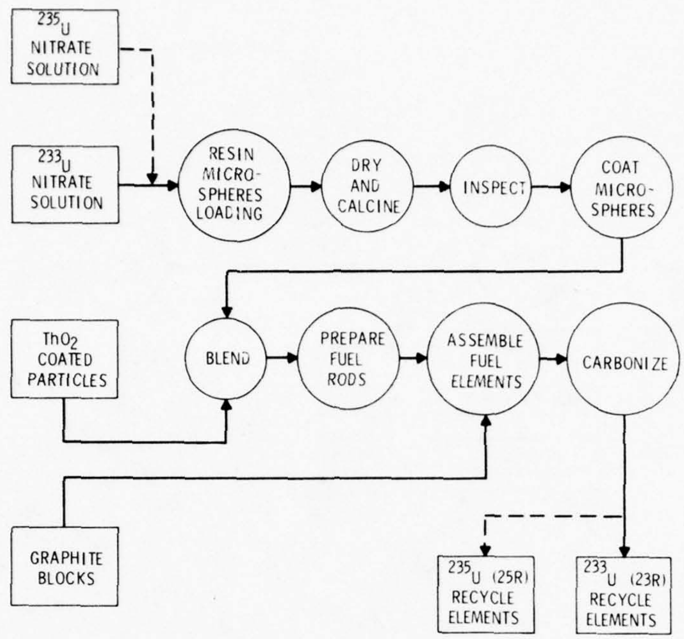


FIGURE 9.2.7. High Temperature Gas-Cooled Reactor Fuel Fabrication

9.3 LIGHT WATER BREEDER REACTOR FUEL CYCLE DESCRIPTION

9.3 LIGHT WATER BREEDER REACTOR FUEL CYCLE DESCRIPTION

Conceptually, an LWBR[13] industry operates on a thorium/uranium-233 fuel cycle which requires the operation of two different reactor cores (prebreeders and breeders) with different fuel cycles. A prebreeder core must produce uranium-233, which does not occur naturally in the earth's crust. Uranium-233 can be produced from thorium-232 during the operation of a prebreeder core, which consumes another fissile isotope, such as plutonium-239 or uranium-235, while producing power for electricity. A conceptual prebreeder fuel cycle based on the consumption of uranium-235 in uranium, which is 10 to 14% enriched, is presented and compared in detail with the conventional LWR fuel cycle. Alternative conceptual prebreeders which utilize plutonium-239 or uranium-235 in 93% enriched uranium are not discussed here.

The main objective of an LWBR industry is to achieve operation of the reactor plants on a self-sustaining breeder fuel cycle which would ultimately use only thorium-232 as fuel. A breeder would operate on a uranium-233/thorium-232 fuel consisting of the following steps:

a) fabrication of uranium-233 and thorium-232 into breeder cores;

b) generation of electricity by heat from fissioning uranium-233 and conversion of some of the thorium-232 into replacement uranium-233;

c) reprocessing of breeder cores to recover uranium-233 and thorium-232 and to remove fission products; and

d) fabrication of the uranium-233/thorium-232 with a small amount of makeup thorium-232 into new breeder cores.

The primary stages for a conceptual breeder fuel cycle are shown in Figure 9.3.1.[14]

9.3.1 Fuel Core Characterization

In the slightly-enriched uranium fueled prebreeder concept presented here, enriched uranium in the form of uranium dioxide pellets (U^EO_2) is placed in small Zircaloy clad rods in their own region within a module (Figure 9.3.2).[15] Thorium is introduced into the module in larger Zircaloy clad rods containing thorium dioxide (ThO_2) pellets. This separation of UO_2 and ThO_2 is necessary in order not to contaminate uranium-233 with uranium-238, uranium-235, or uranium-236, which would reduce the breeding performance. In addition to this separation, the ThO_2 rods are separated from the rest of the module by a Zircaloy can, which allows zoning the water flow up through the module. The fractional water flow by the ThO_2 rods is controlled by an orifice at the bottom of the module. This orifice could be changed as the module is moved during refueling to increase water flow by the ThO_2 rods as the amount of uranium-233 builds and power produced in these rods increases.

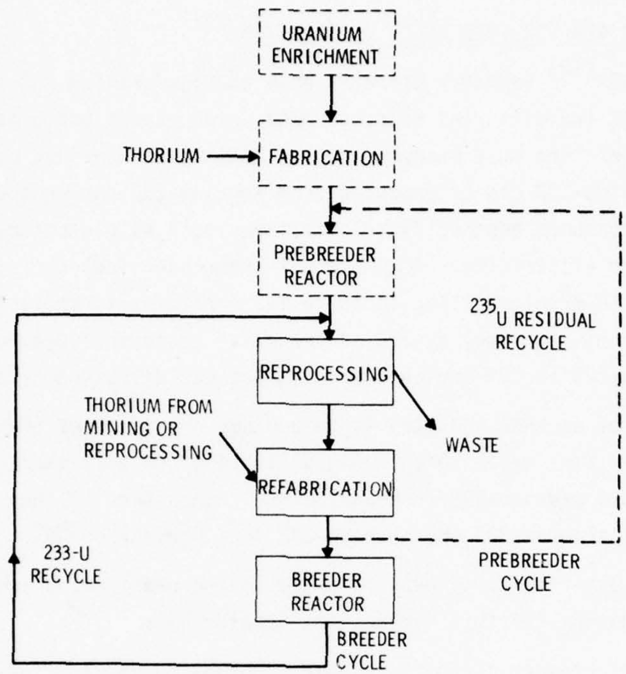


FIGURE 9.3.1. Thorium Fuel Cycle with Light Water Breeder Reactor

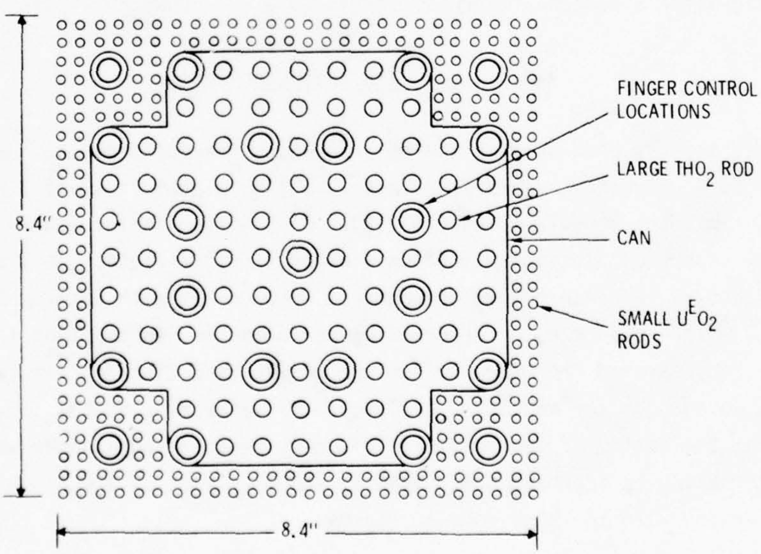


FIGURE 9.3.2. Slightly Enriched Uranium Prebreeder Geometry (Alternative)

An LWBR breeder core concept is similar to the LWBR prebreeder concept, except that the fissile material, uranium-235 and/or plutonium, is replaced by uranium-233 and the breeder concept may use movable fuel control rather than control rods. A typical LWBR breeder fuel assembly may contain seed rods consisting of a binary mixture of uranium-233 oxide and thorium oxide, and thorium oxide axial reflector sections. Blanket rods have a similar configuration, using different percentages of uranium-233 and thorium. A radial reflector blanket consists entirely of thorium dioxide rods.

9.3.2 Light Water Breeder Reactor Fuel Cycle Operation

The discussion of LWBR reprocessing and fuel fabrication that follows is based on the conceptual designs presented in Reference 3, as no such facilities presently exist. As previously mentioned, the fuel cycles for prebreeder and breeder may be significantly different: descriptions are presented for both reprocessing facilities.

9.3.2.1 Reprocessing

Two separate reprocessing plants are conceptualized as model facilities to describe the reprocessing of spent prebreeder fuel. These model plants, which do not exist, are based to a large extent on existing typical reprocessing plants or proven technology and are of the size that would be found in any future LWBR fuel cycle industry. A Purex-type reprocessing plant is used to recover unburned uranium-235 and plutonium produced during irradiation from prebreeder enriched uranium dioxide rods. Products recovered are expected to be converted UF_6 and PuO_2 at the site for further use. To recover the bred uranium and the residual thorium from the prebreeder thoria blanket elements, an Acid-Thorex-type reprocessing plant is expected to be used in another separate plant at the site.

A plant utilizing an Acid-Thorex-type processing scheme is conceptualized as the model facility for processing of LWBR spent breeder fuel. This facility, suffices for all regions of breeder fuel, which differ only in concentrations of fissile (uranium-233) and fertile (thorium) content.

The major facilities constructed at both the prebreeder and breeder reprocessing sites could include the following:

- a fuel receiving and storage facility,
- a mechanical disassembly facility,
- processing buildings housing the requisite reprocessing and waste management equipment,
- radioactive-area ventilation-air-filtration and discharge systems,
- high-level radioactive liquid waste storage,
- offices,
- warehouse and shops,
- steam-generating plants and secondary stacks,
- cooling towers,
- a retention basin,
- product conversion facilities,

- product storage facilities, and
- high-level waste solidification facilities.

9.3.2.2 Purex Plant

For processing the breeder fuel and thoria sections of the prebreeder fuel and HTGR thoria fuel particles, process functions and facility [16,17] descriptions are similar to the Allied-General Nuclear Services' plant, except that a conversion step to produce thorium oxide has been included.

The Purex Plant is similar to that used in LWR fuel reprocessing. Irradiated fuel elements are received at the reprocessing site in shielded casks via rail or truck. Fuel is removed from the shipping casks and stored under water until ready to be reprocessed.

The enriched uranium spent fuel rods are transferred to the Purex separations facility where they are chopped by a shear to expose the core material, then charged directly into a dissolver. The semicontinuous dissolution of the oxide cores minimizes, as well as controls, the peaking of off-gas release. Centrifugation is used to remove any suspended solids in the dissolved feed. The dissolved feed is then introduced into a centrifugal contactor for the first cycle extraction, where uranium and plutonium are separated from bulk fission products.

Pulsed columns are used for separating plutonium from uranium in the first cycle. Plutonium and uranium are processed simultaneously in separate solvent extraction columns. Uranium solutions are given a final silica gel filtration adsorption for removal of any residual zirconium. Final solutions of these plant products are concentrated prior to storage and/or further processing (e.g., to UF_6 and PuO_2). Solvents used in the fuel recycling operation are treated in two parallel solvent treatment systems before reuse.

All aqueous raffinates containing small quantities of fissile material (except solvent treatment wastes and the high activity waste stream) are passed through a recovery extraction system prior to waste treatment operations. All potential fissile containing organic raffinates are recycled through the partitioning column prior to routing to solvent treatment.

The Purex separations plant also includes a plutonium product plant to convert recovered plutonium nitrate to plutonium oxide powder and to provide storage for the product. An oxalate chemical process is used for this purpose. Purex wastes produced in processing irradiated prebreeder uranium fuel elements are, for all practical purposes, identical to those generated in processing irradiated LWR fuel elements and can be managed similarly.

9.3.2.3 The Acid-Thorex/Modified Acid-Thorex Plant

The solvent extraction processing systems for thoria-based fuels consist of five component operations: feed preparation, first cycle extraction, second cycle extraction, uranium salvage and solvent cleanup. Two types of solvent extraction processing could be used to recover the uranium-233 and thorium from the thoria-base core materials: the Acid-Thorex process or the Modified Acid-Thorex process. The effluents from both processes are similar. Figure 9.3.3[18] provides a simplified block diagram of the processes. With the exception of the sorting operation, which is only used for prebreeder fuel, the same flow diagram applies to the reprocessing of breeder fuel and prebreeder thorium-based fuel.

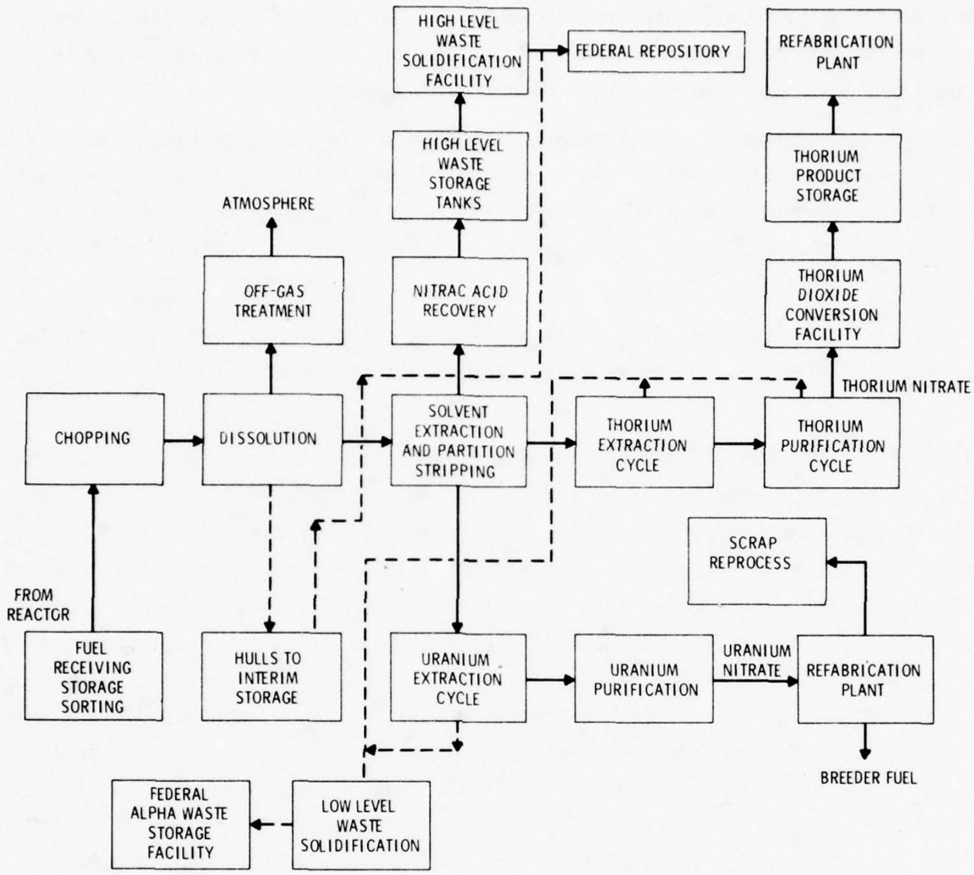


FIGURE 9.3.3. Acid Thorex/Modified Acid-Thorex Reprocessing Facility

Feed solution for these processes is formed by reacting chopped thoria-based element material with a solution containing nitric acid, hydrofluoric acid, alumina nitrate and neutron absorbing material, such as gadolinium or cadmium. As indicated in the description of dissolving thorium oxide fuels for the HTGR, this dissolution step is preliminary. First the addition of fluoride seriously complicates subsequent steps and poses subsequent corrosion problems. Fluoride may form some obnoxious complexes with zirconium, the favored jacketing material for LWRs. Fluoride carry-over into waste solidification is unknown, and may or may not be a problem.

The first extraction cycle for either process serves to separate the uranium and thorium from the bulk of the fission products in the aqueous feed solution. The presence of protactinium-233 and protactinium-231 may pose special problems for processing thorium fuels and wastes. These are briefly discussed in this section.

In the extraction-scrub column, uranium and thorium are extracted into the organic solvent and scrubbed with nitric acid to remove fission products. The scrub solutions and the remainder of the aqueous feed is withdrawn from the bottom of the column, sampled and

transferred to the high-level waste storage tanks, if the uranium concentration meets specifications for future solidification. If the uranium content were above limits, the solution would be recycled through the extraction-scrub column.

The organic solvent phase, containing the thorium and uranium, is withdrawn from the top of the column and transferred to the stripping column, where the thorium and uranium are stripped from the solvent using dilute nitric acid. In the solvent scrub column, the aqueous uranium-thorium solution is concentrated, collected and transferred to the second extraction cycle.

Two alternative systems may be used as the second extraction cycle: the Acid-Thorex Process or the Modified-Acid Thorex Process. Equipment requirements for the two systems are similar.

If further purification is desired for improved quality, the separate thorium and uranium-233 can be processed through an additional solvent extraction cycle; i.e., the thorium can be processed through two cycles of solvent extraction and the uranium-233 through three cycles before being concentrated for load-out. The process waste losses expected from the solvent extraction systems described are estimated from previous thorium processing compaigns. Losses in converting thorium nitrate to thorium oxide are estimated to be 0.1%. For the postulated LWBR process facilities, it is assumed that uranium-233 product losses to waste streams are 0.5% of the initial quantities processed. Thus, the uranium-233 quantities in the associated wastes are comparable with the plutonium content of wastes from plutonium recycle. While the half-life of uranium-233 is seven times longer than for plutonium-239, the alpha emitting daughters of uranium-233 are controlled by a far shorter half-life intermediary (10^4 vs. 10^8 years). Thus, with the greater chemical solubilities of uranium, from a waste hazards standpoint uranium-233 may or may not pose a greater problem than plutonium-239.

Unknown at this time is process complication due to the presence of protactinium in larger quantities than in the relatively small thorium fuel processing campaign completed to date. Of the two major protactinium isotopes involved, protactinium-233 may be most troublesome in processing, while protactinium-231 may lead to long-term waste considerations. Decaying to fissile uranium-233 with a 27-day half-life, protactinium-233 poses potential accountability and criticality problems during processing because of possible plate-out of protactinium on walls or components.

In the future, protactinium-231 may pose a significant long-term waste storage consideration since it is an alpha emitter with a 32,500-year half-life. Unless recycled to uranium-232 or otherwise burned out, protactinium-231 may be present in the wastes in amounts comparable with plutonium-239 quantities in wastes from plutonium recycle. Most protactinium-231 is made two ways: 1) by the n-2n on thorium-232, and 2) by the neutron absorption in thorium-230. The latter is a decay product of uranium-234 and is present in several hundred parts per million in thorium derived from monozite sands, the major thorium source for the world. Substantial thorium reserves in the USA are in the form of thorite ores, which are much lower in thorium-230 content. The use of such ores will reduce the protactinium-231 content, especially in the soft spectrum HTGR types, but considerable amounts will still be made by the n-2n reaction on thorium-232 in the LWBR, which purposely has a hard neutron spectrum to enhance fast fission to assist in breeding.

9.3.3 Light Water Breeder Reactor Fuel Fabrication

The conceptualized LWBR fuel cycle includes both prebreeder and breeder reactor operations. Each of these reactor operations requires a different type of fuel which must be fabricated in separate facilities. Prebreeder fuel is fabricated from enriched uranium-235 dioxide and virgin thorium dioxide. Breeder fuel is composed of uranium-233 dioxide and thorium dioxide which has been recycled from prebreeder and breeder operations. Prebreeder fuel is fabricated using methods similar to the fabrication of conventional LWR fuel, using the same or similar facilities. Breeder fuel, because of its higher radiation level (due to trace amounts of uranium-232 in the uranium-233 produced by the prebreeder) requires shielded fuel fabrication facilities.

A schematic flowsheet depicting process steps for conversion of thorium nitrate to thorium dioxide powder is shown in Figure 9.3.4.[19] The initial process step for conversion

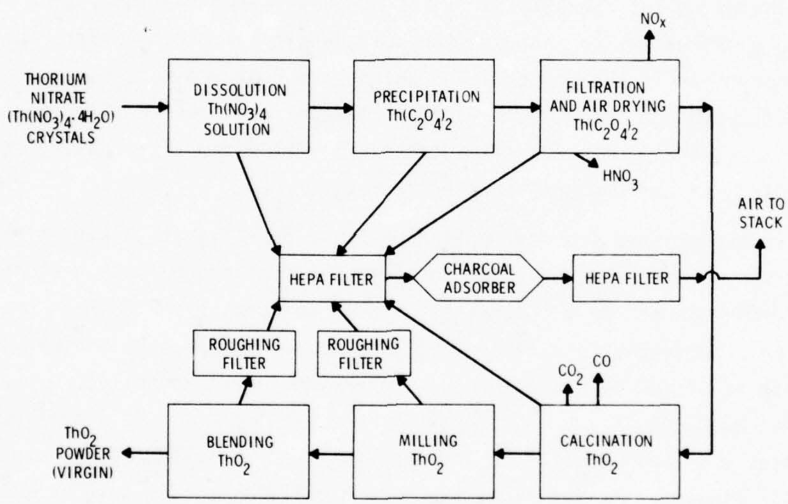


FIGURE 9.3.4. ThO_2 (Virgin) Conversion Process

to thorium dioxide powder is the dissolution of thorium nitrate crystals in acidified water, yielding a thorium nitrate feed solution. The thorium nitrate feed solution is then reacted with oxalic acid to yield a thorium oxalate slurry, which is filtered prior to air drying of the moist filter cake. The filtrate containing small amounts of precipitated thorium and nitric acid is forwarded to the waste treatment facilities.

The thorium oxalate cake is further dried and calcined in an air atmosphere at approximately 430°C (1000°F). This results in its decomposition to thorium dioxide powder. The liberated gases are mainly carbon dioxide with traces of carbon monoxide. The calcined thorium dioxide powder is milled to the desired particle size and blended to ensure a homogeneous mixture inside an enclosure with a roughing filter at the point of discharge. The thorium dioxide powder is fabricated into pellets which are loaded into fuel rods by the mechanical operations, shown in the flowsheet Figure 9.3.5.[20]

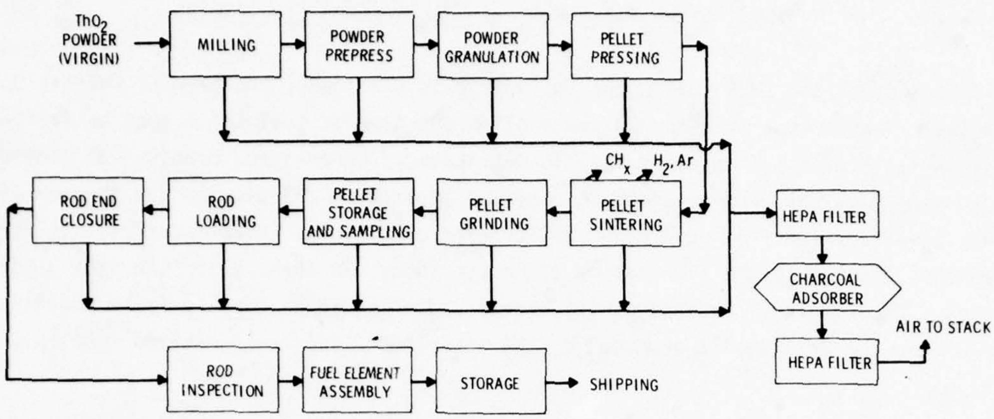


FIGURE 9.3.5. ThO_2 (Virgin) Fuel Fabrication Process.

Thorium dioxide pellet fabrication begins by pulverizing the ThO_2; the powder is then prepressed into short wafers to increase the bulk density of the material and to reduce the amount of entrained air in the powder. The wafers are conveyed to the granulator where the material is granulated and screened to yield a standard agglomerate size for feed to the pellet press. The granulated powder is automatically fed into a die cavity at the pellet press where pellets of uniform density and size are formed.

Pellets are transferred from the pellet pressing operation to a sintering furnace complex where they are sintered to the required density. The sintered pellets are checked for density and dimensions, transferred to a centerless grinder and dry-ground to meet specifications. After inspection, the thorium dioxide pellets are mechanically inserted into fuel rod tubes, one end of which is closed by a welded plug. When the requisite pellets are added, a spring is inserted into each rod, a top end plug is pressed into place, and a seal weld formed. Seal welding completes the fuel rod closure and each fuel rod is inspected for conformance to specifications. Finally, the fuel rods are mechanically assembled into modules of the reactor core and stored until shipment.

For breeder use, the uranium-233 dioxide powder is mechanically blended and milled with recycled thorium dioxide powder in the desired ratios, like the first two steps of the binary pellet fabrication process. The binary powder mixture is fabricated into pellets that are loaded into fuel rods by the mechanical operations, shown in the schematic process flowsheet, Figure 9.3.6.[21] The process for binary pellet fabrication is basically the same as the process steps described for thorium dioxide pellet fabrication. The processes include:

- milling,
- power granulation and pellet pressing,
- pellet sintering,
- dry grinding
- rod loading, and
- fuel element assembly and storage.

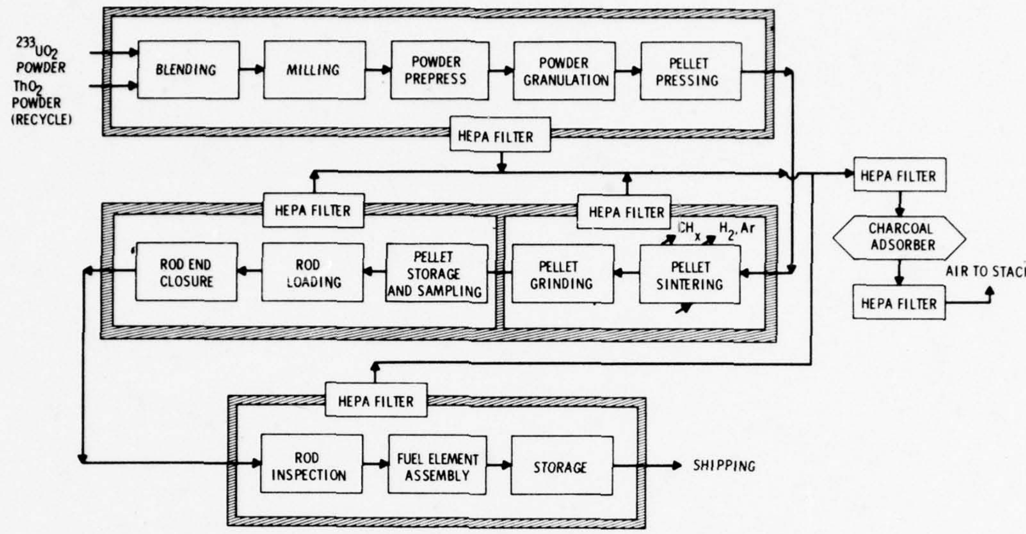


FIGURE 9.3.6. Uranium-235 Dioxide/Thorium Dioxide Fuel Fabrication Process

9.4 THORIUM, U-238, AND ASSOCIATED WASTES

9.4 THORIUM, U-238, AND ASSOCIATED WASTES

Comparison of fission yield curves of Uranium-233 and Plutonium-239 (see Appendix A) indicates a marked increase in yield of masses less than 100, such as krypton, for thorium cycles compared with uranium cycles. The xenon yields are nearly the same. Also a marked decrease is noted in the masses of 100 to 120 for thorium cycles. From a waste standpoint, superimposing the mass yield curves is not sufficient, as the elemental yields for a given mass may be different. Most of the fission product nuclides have short half-lives. The amount of fission products is 5 to 7% greater for a given heat generation by Uranium-233 fission compared with plutonium fission. Moreover, the energy distribution between kinetic particles and gamma rays is less favorable for Uranium-233. This latter effect is not important except in reactors where the moderator cooling is at substantially lower temperatures than the basic coolant, such as in CANDU types. In addition, for thorium fueled LWRs, 5 to 7% more fissile fuel must be consumed because thorium has about 1/5 the fast fission effect of Uranium-238, again increasing the quantities of Uranium-233 fission products. It is not known whether the current initial analyses of Uranium-233/thorium fuels are accounting for such refinements.

Uranium-233 is present in the wastes at levels comparable with plutonium levels in the plutonium recycle fuels. Radiologically, uranium-233 has a 6-times longer half-life, but due to differences in chemistry and daughter activities, the maximum permissible body burdens may be nearly the same.

The thorium fuel cycle is virtually unique in leading to production of protactinium-231 (α - β, γ 32,500 year half-life). This may amount to as much as 1/2% of the waste by weight. This will be less by a factor of two to ten for thorium sources free of thorium-230 (ionium), some U.S. sources of thorium (thorite). However, the world's major thorium sources, monozite sands, have larger thorium-230 concentrations, leading to the larger production of protactinium-231. As the thorium is recycled, thorium-230 is effectively burned out. However, protactinium-231 is still present as a result of the n-2n reaction on thorium-232. The chemistry of protactinium is "ambivalent," thus, it may complicate processing.

The thorium fuel cycle leads to the incidental production of more uranium-232, principally from protactinium-231. Uranium-232 has daughter products that produce significant quantities of alphas and hard gamma rays. However, the half-lives of all daughters are significantly less than the 72-year half-life of uranium-232. Thus, the activity of uranium-232 is a fuel processing and handling problem which includes waste processing, but not a long-term waste storage problem.

Gaseous releases from a facility reprocessing thorium-233 fuel are somewhat greater than those from a slightly enriched uranium reprocessing facility. This is particularly true for krypton, although xenon yields are more nearly equal. Other volatile components will be present in large amounts, such as iodine, ruthenium, etc.

The carbon-14 release from an HTGR reprocessing facility could be up to 15 times larger than that of slightly enriched uranium in LWRs, because of the large amount of graphite in the fuel and the burning operation used to separate the fuel from the structural material.

Contamination of water-cooled reactor coolants with fission products upon fuel element leaks may be considerably less for thorium than for uranium fuels, principally because it is more difficult to dissolve thorium oxide than uranium oxide.

Since uranium-233, and thorium have lower mass numbers than the uranium-235 and uranium-238, a thorium fuel cycle leads to less transuranium waste generated by successive neutron captures. This is augmented in thermal reactors because the capture-to-fission ratio of uranium-233 is lower than for uranium-235 and plutonium-239. However, in the HTGR thorium cycle using uranium-235/uranium-238 enrichment, plutonium may not be recovered in reprocessing. Thus, the plutonium content of the waste could exceed that of LWR waste from which plutonium is removed.

Currently it is necessary to add fluorine to dissolve spent thorium oxide fuel. The effects of fluorine, if any, upon the waste processing and storage are unknown. However, steps could be taken to obviate the fluorine in the processing. This may involve addition of magnesium, calcium, or other elements to thorium oxide which will add to waste volume, but not appreciably to radioactivity. This may, however, increase the solubility of thorium dioxide in water coolant streams, increasing contamination of water coolant streams if fuel jackets develop leaks.

Thorium fuel cycles, even with fissile uranium recycle lead to more uranium-234, thereby radium-226 and thus radon-222 in the wastes, unless plutonium-238 (principally from neptunium-237 is recycled; i.e., fissioned or transmuted. After reactor irradiation, the thorium component requires 15 to 20 years of storage for the incidentally produced thorium-228 to decay to levels found in natural thorium.

The liberated tritium during thorium oxide fuel dissolution may be grossly diluted in the dissolver solution unless a step process is used that can liberate the tritium before the dissolver solution is added. Spent uranium fuels also dilute tritium into the dissolver solution, but alternative processes involving oxidation have been conceived to liberate tritium from uranium dioxide fuels before dissolution. The oxidation process will not work for thorium fuels.

Table 9.4.1 compares the activity of selected isotopes which would be contained in high-level waste if reprocessing were implemented. Activities are shown for: 1) an LWR with partial plutonium recycle, 2) the prebreeder for the LWBR, 3) the LWBR, 4) low-enriched HTGR and 5) highly-enriched LWBR. Not included in these tables is protac-tinium-231 for thorium cycles, which has not been estimated at this time. It appears, however, that protactinium-231 may be present in essentially the same levels as plutonium-239 from plutonium cycles. It would have essentially the same characteristics since it is an alpha emitter with a 32,500-year half-life.

The divisions of Table 9.4.1 are arranged so that the activities of selected isotopes can be compared for the five cases at time intervals of 10, 100, and 1,000,000 years from a reactor discharge. These show only minor differences after 1,000 years, which support a preliminary assessment that as far as fission products are concerned, the high-level waste

TABLE 9.4.1. Comparative Yields for Selected Isotopes in Terms of Ci/GWe·yr

	10 YEARS					100 YEARS					
SELECTED ISOTOPES OR GROUPS	LWR CWMS-3 Cycle	Pre-Breeder Step for LWBR	Breeder Step for LWBR	HTGR LEU	HTGR HEU	LWR CWMS-3 Cycle	Pre-Breeder Step for LWBR	Breeder Step for LWBR	HTGR LEU	HTGR HEU	SELECTED ISOTOPES OR GROUPS
U-232	8.6-1	1.3+4	8.9+4	2.5+3	3.4+3	3.9-1	5.4+3	3.8+4	1.1+3	1.4+3	U-232
U-233*	2.2-3	3.5+3	1.7+4	1.8+3	2.3+3	9.6-3	3.5+3	1.7+4	1.8+3	2.3+3	U-233
U-235	6.4-1					6.4-1					U-235
Pu	4.8+6			1.0+6	2.1+5	2.2+5			5.4+4	7.7+4	Pu
Sr-90	1.9+6	2.0+6	2.3+6	2.3+6	2.5+6	2.1+5	2.1+5	2.4+5	2.5+5	2.7+5	Sr-90
Cs-137	3.2+6	2.2+6	2.3+6	2.4+6	2.7+6	3.8+5	2.7+5	2.8+5	2.9+5	3.0+5	Cs-137
H-3	1.0+4	7.6+3	6.4+3	2.1+4	2.7+4	6.4+1	4.7+1	4.0+1	1.3+2	1.7+2	H-3
Kr-85	1.9+5	1.6+5	2.5+5	8.3+5	3.5+5	5.8+2	4.8+2	7.8+2	2.5+3	1.0+3	Kr-85
I-129	1.4	1.4	2.0	8.0-1	1.0	1.4	1.4	2.0	8.0-1	1.0	I-129
C-14	2.3	12	14	2.5+2	3.2+1	2.2+1	12	14	2.5+2	3.2+1	C-14
Σ Actinide	5.2+6			1.1+6	2.6+5	4.6+5			1.1+5	9.5+4	Σ Actinide
Σ F.P.s	1.1+7			1.0+7	1.0+7	1.2+6			1.1+6	1.2+6	Σ F.P.s
Σ Decay of Heat kW	5.4+1					1.7					Σ Decay of Heat kW

	1,000 YEARS					1,000,000 YEARS					
SELECTED ISOTOPES OR GROUPS	LWR CWMS-3 Cycle	Pre-Breeder Step for LWBR	Breeder Step for LWBR	HTGR LEU	HTGR HEU	LWR CWMS-3 Cycle	Pre-Breeder Step for LWBR	Breeder Step for LWBR	HTGR LEU	HTGR HEU	SELECTED ISOTOPES OR GROUPS
U-232	5.7-5	9.0-5	6.2	1.8-1	2.5-1	--	0	0	--	--	U-232
U-233	1.8-1	3.5+3	1.7+4	1.8+3	2.3+3	5.4+1	5.8+?	2.4+2	3.2+1	3.6+1	U-233
U-235	6.4-1					1.15		?	--	--	U-235
Pu	4.2+4			4.6+3	4.0+2	2.5+1		?	7.8+0	6.0-1	Pu
Sr-90	4.8-5	3.5-5	4.0-5	4.1-5	4.4-5	--	0	0	--	--	Sr-90
Cs-137	1.2+1	2.5-4	2.6-4	6.9+0	5.5+0	0	0	0	0	0	Cs-137
H-3	--	--	0	--	--	--	0	0	--	--	H-3
Kr-85	4.8-23	--	0	--	--	--	0	0	--	--	Kr-85
I-129	1.4	1.4	2.0	0.8	1.0	1.3	1.4	1.9	7.7-1	1.0+0	I-129
C-14	2.0+1	11	13	2.2+2	2.8+1	--	?	0	--	--	C-14
Σ Actinides	1.01+5			2.0+4	5.8+3	9.0+?			6.8+2	9.1+2	Σ Actinides
Σ F.P.s	7.5+2			4.9+2	5.9+2	1.16+?			1.3+2	1.8+2	Σ F.P.s
Σ Decay of Heat kW	3.2					2.09?					Σ Decay of Heat kW

*An example of an isotope (Uranium-233 $-1^5 \times 10^5$ yr) in trace quantities

for the uranium and thorium cycles are essentially equal. This is not as obvious for the actinides which are unique for such systems and may not be equivalent. In addition to the tables of fission products shown, the basic radioactive decay series are presented in Appendix B.

Fuel management schemes greatly affect the character of wastes generated by the thorium fuel cycle. Concentration of transuranics is quite sensitive to yields. Thus, comparison of wastes among alternative fuel cycles is highly dependent upon uncertainties in yield and by strategies adopted to "manage" fuel recycle (see Appendix C).

On the basis of the foregoing somewhat simplified examples, coupled with tabular data shown, one would suspect that the long-term waste, characteristics of the thorium cycle are essentially the same as those for the uranium-238 based cycle. To precisely make such a comparison, the details of all processing and corresponding neutron exposure history must be established. Then all of the isotopes of interest, as well as each fission product, must be followed. Such calculations should be done for all fuel cycles of interest. Moreover, studies should include the scope of likely fueling strategies and likely yields which can have a large impact on the quantities and character of actinide wastes.

Internal management directives of DOE require that uranium-233 contaminated waste be handled in the same manner.

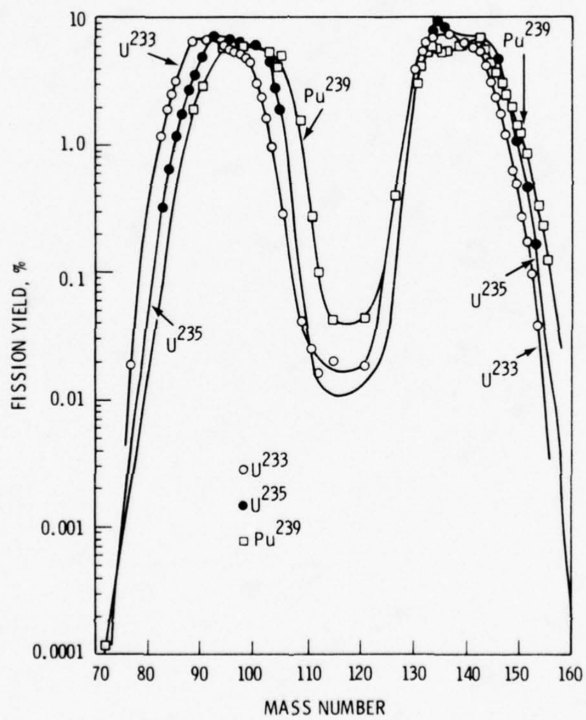
APPENDIX 9A

FISSION YIELD CURVES

APPENDIX 9A

FISSION YIELD CURVES

Superimposed plots can be used to compare the fission yield from uranium-233, uranium-235, and plutonium-239 in thermal neutron reactors. Correspondingly, uranium-238 yield curve in LWRs is for fast fission rather than thermal spectrum neutrons and falls within these three. The superimposed fission yield curves indicate that for mass numbers greater than 120, the yield for all heavy, high yield isotopes is nearly the same (within a factor of 2); but for mass less than 120, the lighter the specie fissioned, the yield curve shifts generally towards light mass isotopes. This is reflected in the marked increase in yield of krypton for uranium-233 cycles compared with plutonium cycles, yet the xenon yields are nearly the same. Also, this leaves a marked decrease in the masses of 120 to 100 for uranium-233 cycles. From a waste standpoint, superimposing the mass yield curves is not sufficient as the elemental yields for a given mass may be different. However, most of the fission product nuclides have short half-lives.



APPENDIX 9B

RADIOACTIVE DECAY SERIES

APPENDIX 9B

RADIOACTIVE DECAY SERIES

The naturally occurring radionuclides can be divided into those that occur singly and those that are components of three distinct chains of radioactive elements. The chain of naturally occurring radioactive elements are:

- the uranium series which originates with uranium-238 [4n + 2) series];
- the actinium series which originates with uranium-235 [4n + 3) series];
- and the thorium series which originates with thorium-232 (4n series).

These three families of radioactive heavy elements are all found in the earth's crust. A fourth heavy element series, the neptunium series [(4n + 1) series], includes uranium-233, but does not exist in nature.

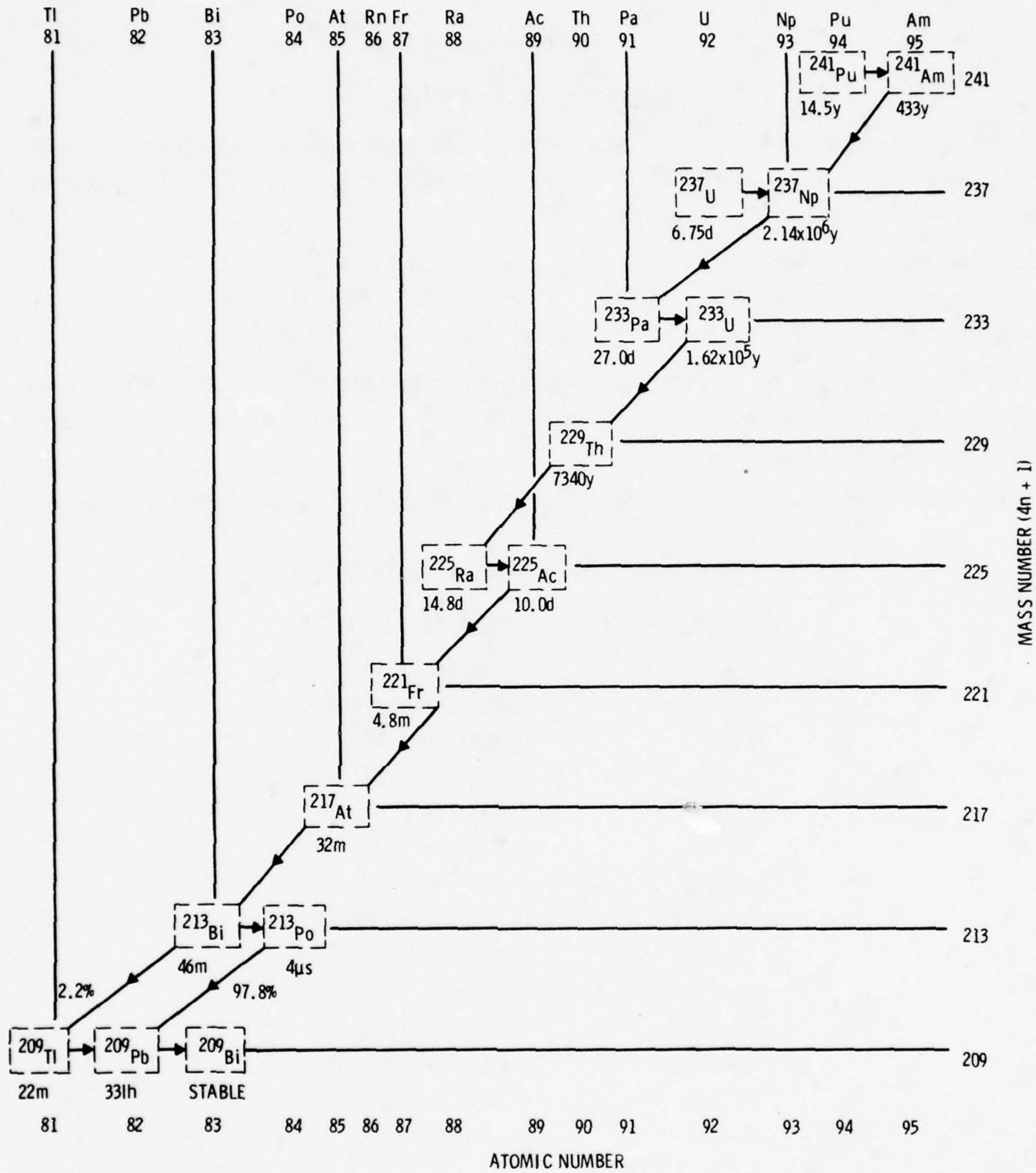
Fueling isotopes of interest are listed in Table 9B-1 with the corresponding series.

TABLE 9B-1. Radioisotope Decay Series

Fueling Isotope	Corresponding Series
Pu-239	U-235 (4n + 3)
Pu-240	U-232-Th-232 (4n)
Pu-241	U-233 (4n + 1)
Pu-242	U-238 (4n + 2)
U-233	U-233 (4n + 1)
U-234	U-238 (4n + 2)
U-235	U-235 (4n + 3)
U-236	U-236-Th-232 (4n)
U-238	U-238 (4n + 2)
Pu-238	U-238 (4n + 2)
Np-237	U-233 (4n + 1)
U-232	U-232-Th-232 (4n)
Pa-231	U-235 (4n + 3)
Th-232	Th-232 (4n)

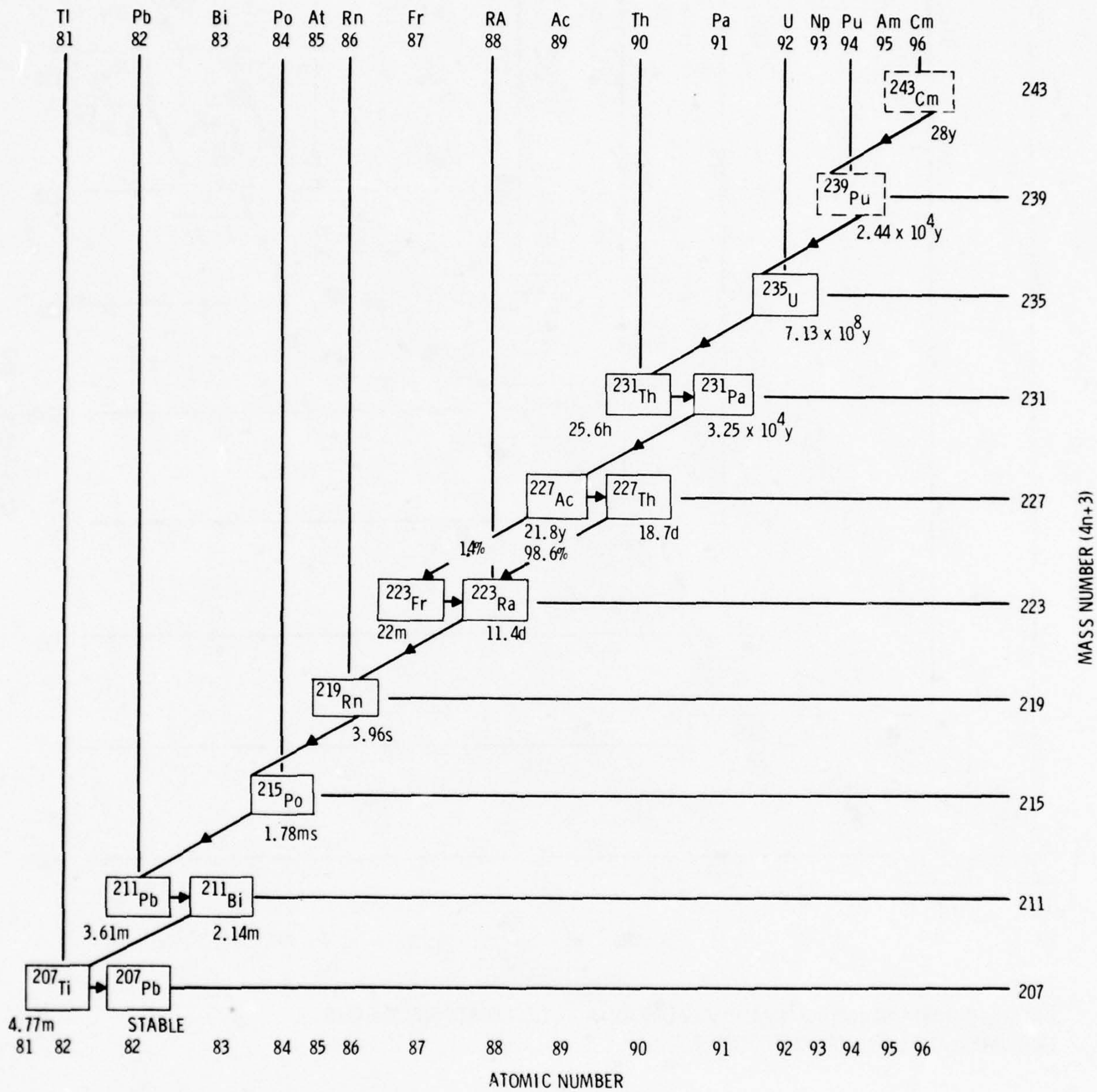
The decay chains of the four heavy element decay series are exhibited diagrammatically in Figures 9B-1 to 9B-4. The members of each chain are shown on a two-dimensional representation. The half-life of each member nuclide is exhibited with the individual boxes. For special emphasis the decay of uranium-232 is shown in Figure 9B-5.

Within these chains are some of the isotopes generated in thorium-based fuel cycles that may require specific consideration for long-term waste storage. These include protactinium-231 and plutonium-238; the latter because plutonium may not be recycled in these instances.



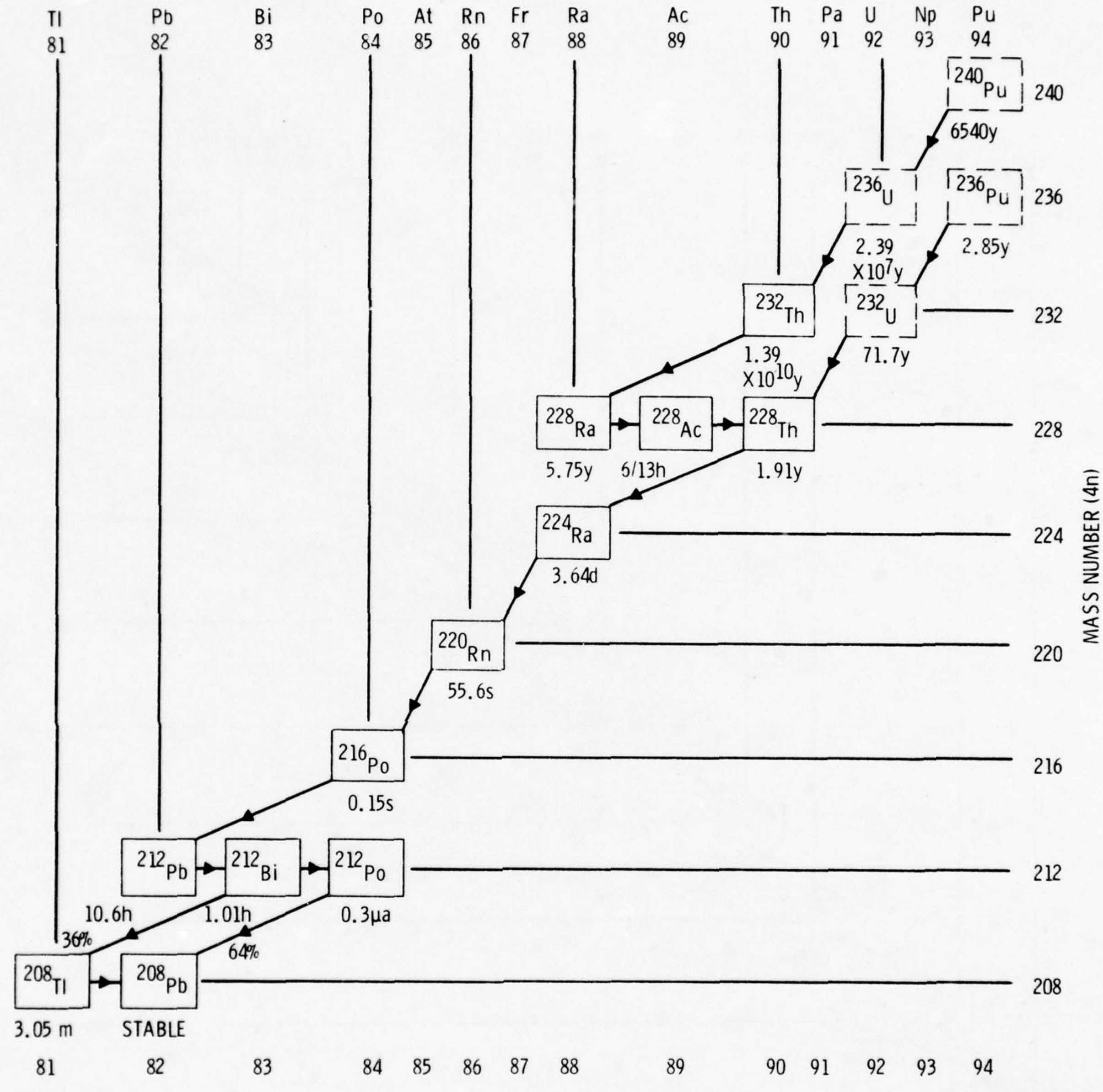
THE SOLID BOXES DENOTE NATURALLY OCCURING NUCLIDES. DIAGONAL ARROWS INDICATE α AND HORIZONTAL ARROWS β DECAY.

FIGURE 9B-1. Uranium-233 (4n + 1) Series



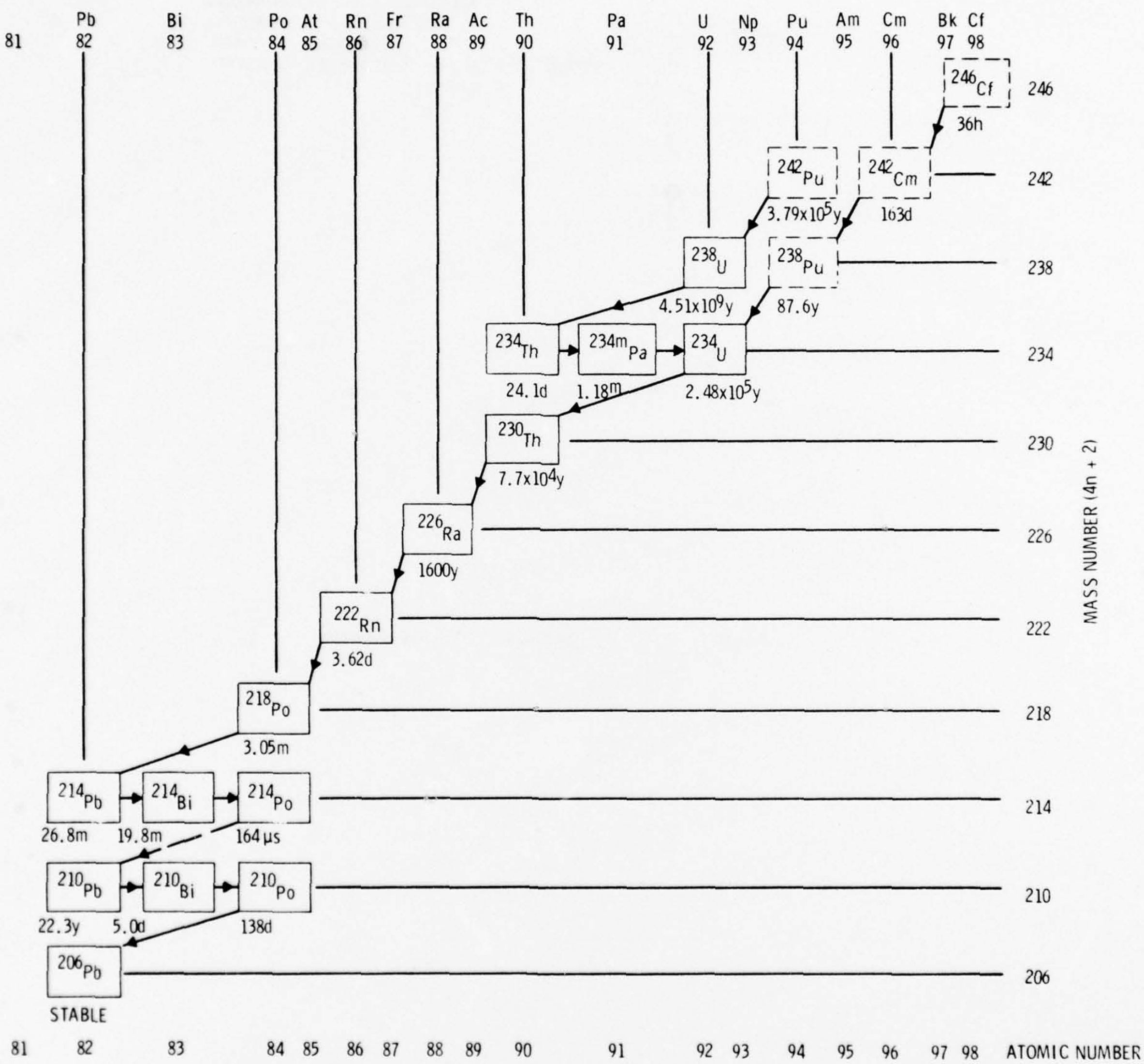
THE SOLID BOXES DENOTE NATURALLY OCCURRING NUCLIDES. DIAGONAL ARROWS INDICATE α AND HORIZONTAL ARROWS β DECAY. CHAIN BRANCHES OF < 1% OMITTED.

FIGURE 9B-2. Uranium-235 (4n + 3) Series



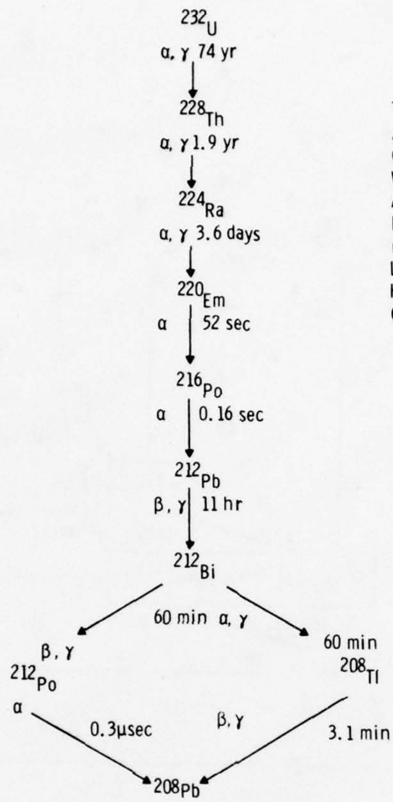
THE SOLID BOXES DENOTE NATURALLY OCCURRING NUCLIDES DIAGONAL ARROWS INDICATE α AND HORIZONTAL ARROWS β DECAY.

FIGURE 9B-3. Uranium-236 (4n) Series



THE SOLID BOXES DENOTE NATURALLY OCCURRING NUCLIDES. DIAGONAL ARROWS INDICATE α AND HORIZONTAL ARROWS β DECAY. CHAIN BRANCHES OF < 1% OMITTED.

FIGURE 9B-4. Uranium-238 (4n + 2) Series



THE GAMMA OF T1-208 IS 2.6 Mev AND LEADS TO THE ONEROUS CHARACTERISTICS ASSOCIATED WITH U-232. U-232 DECAY GENERATES 4.4 WATTS/GRAM, A RELATIVELY SIGNIFICANT AMOUNT IN VIEW OF ITS 71-YEAR HALF-LIFE. (Pu-238 WITH AN 89-YEAR HALF-LIFE GENERATES .56 WATTS/GRAM.)

FIGURE 9B-5. Uranium-232 Decay Scheme

APPENDIX 9C

FUEL CYCLE MANAGEMENT

APPENDIX 9C

FUEL CYCLE MANAGEMENT

The concentration, as well as composition, of transuranics in fission product wastes is highly dependent upon the fuel management schemes employed. Currently, the nuclear industry is working within a narrow band of possible nuclear fueling schemes and, more particularly, of possible fuel management within a given scheme. As conditions change, there will be incentives to alter fuel management, as well as to employ different fueling systems. The following sections briefly indicate how much variance in composition and concentration of the transuranics would be possible within fuel management of plutonium and uranium-233 recycle systems. Except for concentration, little variance will be noted in the composition of fission products as a result of employing different fuel management schemes; thus, fission products are not mentioned in this section. Variations in fuel management can be done purposely to reduce the concentration of transuranics in the wastes. Certainly it will become progressively more costly to reduce the concentration of transuranics in the wastes; however, at nominal cost, new technology might reduce these transuranics from presently accepted levels.

C.1 FUEL CYCLE MANAGEMENT ALTERNATIVES AND WASTES

In the instance of recycle of the fissile-fertile chains developed by irradiation of thorium and uranium-238, there is a great leverage on the concentration of the higher isotopes by the yields throughout all steps. This can be appreciated qualitatively for uranium-236 in the case of thorium-232, and plutonium-242 in the case of uranium-238, by the following example for the plutonium series.

During irradiation of uranium-238, plutonium isotope concentrations approach equilibrium levels. The first isotope to come near equilibrium is that of plutonium-239; its equilibrium level is that level at which its formation rate (essentially determined from uranium-238) equals its destruction rate, through fission, transmutation, decay, and losses. This is sequentially followed by plutonium-240, plutonium-241, and plutonium-242, each coming to equilibrium. Approximate equations for the equilibrium concentrations are:[a]

$$^{239}\mathrm{Pu\ concentration} = \left(^{238}\mathrm{U\ concentration}\right)\left(\frac{\alpha a^{241}\mathrm{Pu}}{\hat{\sigma} a^{239}\mathrm{Pu} + \mathrm{normalized\ decay}}\right)\left(\mathrm{Normalized\ Yield}\right) \quad (1)$$

a. These equations simplify normalized processing yield and decay to the extent that the uranium-238 density is considered constant. For the burnup anticipated in LWRs, this is adequate for cursory planning; however, virtually all reactor burnup codes in use today rigorously account for the discrete yield, and continuity for the destruction of uranium-238, its decay, and how it affects the equilibrium levels of plutonium isotopes.

$$^{240}Pu \text{ concentration} = \left(^{239}Pu \text{ concentraton}\right)\left(\frac{\hat{\sigma}c\,^{239}Pu}{\hat{\sigma}a\,^{240}Pu + \text{normalized decay}}\right)\left(\text{Normalized Yield}\right) \quad (2)$$

$$^{241}Pu \text{ concentration} = \left(^{240}Pu \text{ concentration}\right)\left(\frac{\hat{\sigma}c\,^{239}Pu}{\hat{\sigma}a\,^{241}Pu + \text{normalized decay}}\right)\left(\text{Normalized Yield}\right) \quad (3)$$

$$^{242}Pu \text{ concentration} = \left(^{241}Pu \text{ concentration}\right)\left(\frac{\hat{\sigma}c\,^{241}Pu}{\hat{\sigma}a\,^{242}Pu + \text{normalized decay}}\right)\left(\text{Normalized Yield}\right) \quad (4)$$

(the above expressions are highly simplified, particularly regarding normalizations)

It should be noted that the yield term enters the higher order isotope equilibrium levels to a progressively large extent.

$$^{242}Pu \text{ concentration} = {}^{238}U \text{ concentration (yield)}(^{239}Pu \text{ concentration})(\text{yield})(^{240}Pu \text{ concentration})(\text{yield})(^{241}Pu \text{ concentration})(\text{yield}) \quad (5)$$

(a highly simplified relationship.)

The yield terms are the same for all isotopes (unique radioactive decay is in the denominators of each concentration term). Thus, plutonium-242 concentration is equal to the constant times the yield to the fourth power. Thus:

$$^{242}Pu \text{ concentration} = K_1 \text{ (yield)}^4; \quad (6)$$

and similarly,

$$^{236}U \text{ concentraton} = K_2 \text{ (yield)}^4 \quad (7)$$

The yield terms are usually considered to be the yield from chemical processing and run about 0.99. For all of the plutonium isotopes, the amount that may go to wastes is proportionate (1-yield). Thus, if the yield is 0.98 rather than 0.99, the plutonium waste streams are increased by 100%. At 0.99 yield, the plutonium-242 concentration in the reactor fuel is about 0.98 of possible. Since plutonium-242 is a parasite in thermal reactor spectrums, reducing its concentration in recycle fuel is desirable. Thus if one sets aside 10% of the plutonium, then the plutonium-242 equilibrium is 0.66 of the theoretical yield; if 20%, 0.41; if 30%, 0.22. Of course, the plutonium returned to the reactor beings to approach equilibrium again, but the plutonium-242 concentration takes sufficiently longer (Plutonium-239, plutonium-240, and plutonium-241 must reach equilbrium first) that this allows the plutonium

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a. The plutonium set aside must, of course, be burned without the presence of uranium-238 and, except for plutonium-242 concentration effects, the plutonium-242 must be burned in neutron flux spectrums more favorable to it than that produced in concentrated loads in LWRs (which was the only mode examined in the GESMO studies). Methods of burning plutonium-242 in a more useful fashion are included in other studies.

in the presence of uranium-238 to have substantially higher reactivity if the percentage set aside is properly chosen.[a]

C.2 RAMIFICATIONS OF THORIUM CYCLES TO WASTE MANAGEMENT

Similarly, uranium-236 build-up in uranium-233 systems can be "managed," and if efforts to breed with the thorium system are pursued, minimizing the uranium-236 content will be pursued in the long run where successive recycles are employed. Isolating uranium-236 for incorporation into wastes without further irradiation may well be chosen since it has a long half-life. However, unless isotopically separated, uranium-236 batches will contain values of uranium-233 and uranium-235; thus, some burnout will be considered. This in turn will produce neptunium-237, and thereby plutonium-238, which may not be recycled in a thorium system. If so, plutonium-238 will lead to uranium-234 contents that are of possible long-term concern, in view of its decay to radon-222 controlled by the half-life of uranium-234 of 2.48×10^5 years.

The important consideration is that the concentration of transuranics is even more sensitive to yields [roughly proportionate to 1-yield)]. Thus, the precision of comparisons of wastes among alternative fuel cycles is dominated by uncertainties in yield and by strategies adopted to "manage" fuel recycle. The foregoing arguments apply for managing all of the successive neutron absorption chains of economic interest.

There are other actinide series of significance when comparing the uranium-238 and thorium systems. These include protactinium-231 generated in the thorium cycle and uranium-236 from the slightly enriched uranium cycle. The concentration in the wastes of protactinium-231 with its 32,500-year half-life will depend on how it is managed in the successive recycle. There is, of course, an incentive to recycle the protactinium in the thorium cycle to assure that the protactinium-233 fully decays to uranium-233; and yet by so doing, protactinium-231 is not "burned out," its concentration in the wastes is correspondingly increased to levels that will approach the plutonium-239 concentration in wastes from plutonium recycle. Either of these alternatives could be avoided by purposefully irradiating protactinium-231 as an isolated target and by adding uranium-232 into the high-level wastes in dilutions so localized heating will not be produced.

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a. These equations simplify normalized processing yield and decay to the extent that the uranium-238 density is considered constant. For the burnup anticipated in LWRs, this is adequate for cursory planning; however, virtually all reactor burnup codes in use today rigorously account for the discrete yield, and continuity for the destruction of uranium-238, its decay, and how it affects the equilibrium levels of plutonium isotopes.

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20. See Reference 3, p. IX.G.6-17.

21. See Reference 3, p. IX.G.6-77.

ACRONYMS LIST

ACRONYMS LIST

A-E	architect-engineer
AAPG	American Association of Petroleum Geologists
ACVSF	air-cooled vault storage facility
AEC	Atomic Energy Commission
AECL	Atomic Energy of Canada, Limited
AFR	away from reactor (spent fuel storage)
AGNS	Allied General Nuclear Services
ALARA	as low as reasonably achievable
AMAD	aerodynamic median activity diameter
AP	activation product
API	American Petroleum Institute
APS	atmospheric protection system
BFRSS	Barnwell Fuel Receiving and Storage Station
BIF	bitumen immobilization facility
BPPF	Barnwell Plutonium Product Facility
BTU	British thermal unit
BWR	boiling water reactor
CANDU	Canadian heavy water reactor
CDC	canister decontamination cell (cubicle)
CFR	Code of Federal Regulations
CIF	cement immobilization facility
CRWM	Committee on Radioactive Waste Management
CUP	cask unloading pool
CVCS	chemical and volume control system
CW	canistered waste
CWMS	Generic Environmental Impact Statement on Commercial Radioactive Waste Management, DOE-1559
CWTF	cask weld test facility
DCSF	dry caisson storage facility
DF	decontamination factor
DOE	Department of Energy
DOG	dissolver off-gas
DOP	dioctyphthalate
DOT	Department of Transportation
DTPA	diethylenetriamine pentaacetic acid
ECWS	essential cooling water system
EPC	engineering, procurement, and construction
ER	environmental report
ERDA	Energy Research and Development Administration
ESFS	engineered safety features systems
ESPS	essential spray pond system
FFTF	Fast Flux Test Facility
FP	fission product
FPF	fuel packaging facility
FRP	fuel reprocessing plant
FRPF	fuel residue packaging facility
FRSSF	fuel residue subsurface storage facility
FRVSF	fuel residue vault storage facility
FRW	fuel residue waste
FSA	fuel storage area
FSAR	Final Safety Analysis Report
FSB	fuel storage basin
FTF	fuel transfer facility
FTP	fuel transfer platform
GEIS	Generic Environmental Impact Statement
HCF	hulls compaction facility
HEPA	high-efficiency particulate air (filter)
HEU	highly enriched uranium
HLLW	high-level liquid waste
HLW	high-level waste
HM	heavy metal
HMA	hot maintenance area
HMF	hulls melting facility
HPF	hulls packaging facility
HTD	hulls transfer device
HTGR	high temperature gas-cooled reactor
HVAC	heating, ventilation, and air conditioning
IAEA	International Atomic Energy Agency
IBC	in-bed combustion
ICPP	Idaho Chemical Processing Plant
IFSF	independent fuel storage facility
IIPSF	independent interim plutonium oxide storage facility

ILLW	intermediate-level liquid waste
ILW	intermediate-level waste
INEL	Idaho National Engineering Laboratory
IPSF	interim plutonium oxide storage facility
ISFS	independent spent fuel storage
ISFSB	independent spent fuel storage basin
ISFSF	independent spent fuel storage facility
LAA	limited access area
LEU	low-enriched uranium
LHD	load-haul-dump
LLW	low-level waste
LN_2	liquid nitrogen
LSA	low specific activity
LWBR	light water breeder reactor
LWR	light water reactor
M&M	men and materials
MFBM	thousand board feet measure
MFRP	General Electric Company's Midwest Fuel Reprocessing Plant
MOX FFP	mixed oxide fuel fabrication plant
MP	mine production
MSRE	molten salt reactor
MTHM	metric ton heavy metal
NAA	normal access area
NAC	Nuclear Assurance Corporation
NAS	National Academy of Sciences
NASA	National Aeronautics and Space Administration
NFS	Nuclear Fuel Services
NHLSW	non-high-level solid waste
NLI	National Lead Industries
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
NWTS	National Waste Terminal Storage
ORIGEN	a computer program to calculate isotopic composition of irradiated nuclear fuel
ORNL	Oak Ridge National Laboratory
ONWI	Office of Nuclear Waste Isolation
OWI	Office of Waste Isolation
P-T	partioning and transmutation
PCWS	plant cooling water system
PFRF	packaged fuel receiving facility
PNL	Pacific Northwest Laboratory
POG	process off-gas
PSAR	preliminary safety analysis report
PWR	pressurized water reactor
R&D	research and development
RAA	restricted access area
RBOF	receiving basin for offsite fuel, Savannah River Plant
RCS	reactor coolant system
SCRA	storage cask receiving area
SCSF	surface cask storage facility
SF	spent fuel
SFPF	spent fuel packaging facility
SFRSS	spent fuel receiving and storage station
SFSF	spent fuel storage facility
SHLW	solidified high-level waste
SNM	special nuclear material, i.e., enriched uranium and plutonium
SRP	Savannah River Plant
SSC	sealed storage cask
SSCF	sealed storage cask facility
TBP	tributyl phosphate
TD	theoretical density
TN	Transnuclear Inc.
TRU	transuranic
TSA	transuranic storage area
TWCA	Teledyne Wahchang Albany
U-F	urea-formaldehyde
VE	ventilation exhaust
VOG	vessel off-gas
WBS	water basin storage
WBSF	water basin storage facility
WBSF-PF	water basin storage facility for packaged fuel
WCC	waste calcination cell (cubicle)
WCF	waste calcination facility
WIPP	Waste Isolation Pilot Plant
WTEB	waste tank equipment building
WVC	waste vitrification cell
WVF	waste vitrification facility

MEASUREMENT UNITS AND CONVERSIONS

MEASUREMENT UNITS AND CONVERSIONS

This report preferentially uses the metric system of measurements as defined by the International System of Units (SI). Common English units are often also included in parentheses. Prefixes used with the metric units are defined as follows:

Prefix	Abbreviation	Factor
giga	G	10^9
mega	M	10^6
kilo	k	10^3
centi	c	10^{-2}
milli	m	10^{-3}
micro	μ	10^{-6}
nano	n	10^{-9}

The following lists identify the symbols used in this report and the factors for converting between the SI and English units.

Symbols for metric units used in this report are:

Symbol	Name
°C[a]	degree Celsius
d[a]	day
g	gram
h (or hr)	hour
ha	hectare
kWh	Kilowatt-hour
J	joule
ℓ	liter
m	meter
min	minute
<u>M</u>	gram-mole/liter
MT	metric ton
MW-hr (or MWh)	megawatt-hour
s (or sec)	second
W	watt

a. Units which are not strictly SI but which are widely used.

Symbols for other units used in this report are:

Symbol	Name
atm	atmospheric pressure
BTU	British thermal unit
Ci	curie
°F	degree Fahrenheit
ft	feet
gal	gallon
in.	inch
lb	pound
MFBM	thousand board feet measure
psi	pounds/square inch
R	roentgen
rem	roentgen equivalent man
yd	yard
yr	year

To convert metric to English, multiply by:

Metric	English	Factor
°C	°F	(°C x 9/5) + 32
cm	inch	0.3937
ha	acre	2.47
kg	lb	2.205
km	mile	0.6214
ℓ	gal	0.2642
m	ft	3.281
m^2	ft^2	10.76
m^3	MFBM	0.424
m^3	ft^3	35.31
m^3	gal	264.2
m^3	yd^3	1.308
MT	ton	0.9070
W	BTU/hr	3.413
W-s/kg-°C	BTU/lb-°F	2.39×10^{-4}
W/m-°C	BTU/hr-ft-°F	0.576

To convert English to metric, multiply by:

English	Metric	Factor
acre	ha	0.405
BTU	W-hr	0.2931
BTU/lb-°F	W-s/kg-°C	4187
BTU/hr-ft-°F	W/m-°C	1.735
°F	°C	(°F-32) x 5/9
ft	m	0.3048
ft^2	m^2	0.0929
ft^3	m^3	0.0283
gal	ℓ	3.785
gal	m^3	3.785×10^{-3}
inches	cm	2.540
lb	kg	0.4536
mile	km	1.609
MFBM	m^3	2.360
ton	MT	1.103
yd^3	m^3	0.7646